Technical Evaluation Report

Draft Waste Incidental to Reprocessing Evaluation for Closure of Waste Management Area C, Hanford Site, Washington

Final Report

U.S. Nuclear Regulatory Commission
Office of Nuclear Material Safety and Safeguards
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Table of Contents

LIST OF FIGURES

Figure		Page

LIST OF TABLES

Table | Page

ABBREVIATIONS/ACRONYMS

ADAMS	Agencywide Document Access and Management System
AEA	Atomic Energy Act of 1954, as amended
ALARA	As Low As Is Reasonably Achievable
Am	americium
Ba	barium
BBI	Best Basis Inventory
BBIM	Best Basis Inventory Management
Bq	becquerel
C	carbon
°C	degree Celsius
CCU	Cold Creek unit
CCMS	Camera Computer-Aided Design Modeling System
CMS	Corrective Measures Study
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
COC	contaminants of concern
Cs	cesium
CPGWM	Central Plateau Groundwater Model
D2EHPA	di-(2-ethylhexyl) phosphoric acid
DCF	dose conversion factor
DOE	U.S. Department of Energy
DOE-ORP	U.S. Department of Energy, Office of River Protection
DOE-RL	U.S. Department of Energy, Richland Operations Office
DOT	U.S. Department of Transportation
DST	double-shell tank
EB	engineered barrier
EDS	energy-dispersive spectroscopy
EDTA	ethylene diamine tetraacetic acid
EHM	equivalent homogeneous medium
EM	Environmental Management
EMCF	Environmental Model Calculation Files
EMMA	Environmental Model Management Archive
EPA	U.S. Environmental Protection Agency
ERDF	Environmental Restoration Disposal Facility
ET	evapotranspiration
Eu	europium

ABBREVIATIONS/ACRONYMS (continued)

°F	degrees Fahrenheit
FEP	features, events, and processes
FFTF	Fast Flux Test Facility
FR	Federal Register
FS	feasibility study
ft	foot
ft^3	cubic feet
FY	fiscal year
g	grams
gal	gallon
GTCC	Greater-Than-Class-C
^3H	tritium
H3	Hanford H3 gravel formation
HDW	Hanford Defined Waste or Hanford Defined Waste Model
HEDTA	hydroxyethylene diamine triacetic acid
HFFACO	*Hanford Federal Facility Agreement and Consent Order* (Tri-Party Agreement)
HLW	high-level radioactive waste
HPB	Hanford Prototype Barrier
hr	hour
HTWOS	Hanford Tank Waste Operations Simulator
I	iodine
IA	Interagency Agreement
ICRP	International Commission on Radiation Protection
ID	Idaho
in	inches
K_d	distribution coefficient
K_h	hydraulic conductivity
km	kilometers
L	liters
LAW	low-activity waste
LHS	Latin Hypercube Sampling
LLW	low-level radioactive waste
m	meters
m^3	cubic meters
MBq	mega becquerel
MCi	mega curies
mi	miles
mL/g	milliliters/gram
mm	millimeters
MOU	Memorandum of Understanding
mph	miles per hour
mrem	millirem
mSv	millisievert
Nb	niobium

ABBREVIATIONS/ACRONYMS (continued)

nCi	nano Curie
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
Perma-Fix	Perma-Fix Environmental Services, Inc.
PFNW	Perma-Fix Norwest Richland, Inc.
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
POC	point of calculation
Pu	plutonium
PUREX	Plutonium Uranium Extraction Plant
QA	quality assurance
QC	quality control
RAI	request for additional information
RCA	RCRA Closure Analysis
RCRA	Resource Conservation and Recovery Act of 1976
REDOX	Reduction-Oxidation (Plant)
rem	unit of dose equivalent
RF	Ringold formation
RFI	RCRA Facility Investigation
RI	Remedial Investigation
RSD	relative standard deviation
s	second
Sb	antimony
SC	South Carolina
SEM	scanning electron microscopy
SOF	sum-of-fractions
Sr	strontium
SRM	Staff Requirements Memorandum
SRP	standard review plan
SST	single-shell tank
STOMP©	Subsurface Transport Over Multiple Phases
Sv	Sievert
SZ	saturated zone (also the aquifer)

ABBREVIATIONS/ACRONYMS (continued)

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
TWINS	Tank Waste Information Network System
U	uranium
UPR	unplanned release
USGS	U.S. Geological Survey
UZ	unsaturated zone (also the vadose zone)
VZ	vadose zone (also the unsaturated zone)
WA	Washington
WAC	Washington Administrative Code
WCS	Waste Control Specialist LLC
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 has requested that the NRC provide independent technical advice and consultation regarding DOE WIR determinations for closure of the HLW storage tanks in Waste Management Area C (WMA C) and other potential tasks at the Hanford Site.[1]

In accordance with this IA, the DOE provided a Draft WIR Evaluation for WMA C (and supporting documents, including a Performance Assessment) to the NRC, requesting the NRC's consultative technical review to determine if the Draft WIR Evaluation meets the DOE Manual 435.1-1 criteria for WMA C WIR to be managed as LLW.

In the Draft WIR Evaluation, DOE addressed the waste residuals which remain in the WMA C waste tanks and ancillary structures at the time of WMA C closure. Closure of the individual single-shelled tanks and WMA C in its entirety would occur in three major steps:

1) Retrieval of waste in the tanks,
2) Filling the tanks with grout for stabilization, and
3) Placement of an engineered surface cover barrier.

The final state of a tank farm that is considered in the performance assessment is, therefore, a set of underground grouted tanks with associated ancillary equipment containing residual

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

wastes (after waste retrieval is completed), covered by a modified Resource Conservation and Recovery Act of 1976 (RCRA) Subtitle C engineered surface cover.

The NRC staff conducted an independent, risk-informed, technical review of the Draft WIR Evaluation for WMA C, using the risk insights 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 used DOE's models as well as independent calculations to risk-inform its review.

The Draft WIR Evaluation for WMA C, and the NRC staff review included in this TER, do not address other facilities or systems, waste removed from the waste tanks and ancillary structures and disposed elsewhere, or the contaminated soil and groundwater from previous leaks or releases at WMA C. In addition, the NRC staff review did not address the question of whether waste residuals should be managed as TRU waste, since DOE did not seek this type of technical advice in the IA requesting consultation from the NRC.

The NRC staff's review results and recommendations for the Draft WIR Evaluation for WMA C and supporting documents are provided for Criteria A, B, and C in Sections 2, 3, and 4, 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 area in the Draft WIR Evaluation for WMA C (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 provided in each subsection are collated into Table 5-1, which provides a listing of all the recommendations in this TER from Sections 2, 3, and 4. The recommendations are categorized as (1) applicable to the Draft WIR Evaluation for WMA C, (2) consider for future evaluations for other waste management areas, or (3) general technical recommendations that would generally improve the basis for the technical information but are not essential to the evaluation for WMA C. Recommendations that are categorized as "consider for future evaluations" could be risk-significant depending on the specific details of the waste management area being evaluated. The "general technical recommendations" are simply noted as best practices for performing waste evaluations.

The NRC staff provides many technical recommendations in this TER, as presented in Table 5-1, however, these recommendations do not change the conclusions of this TER with respect to meeting the DOE Manual 435.1-1 criteria, as summarized below. Most of the recommendations are categorized as consider for future evaluations or general technical recommendations and not the current WMA C WIR evaluation because of the low projected risks from WMA C waste residuals. However, these recommendations are relevant to future waste evaluations and waste determinations, especially for waste management areas with higher projected risks, and could serve to enhance the technical basis for the Draft WIR Evaluation for WMA C and decrease uncertainties.

The NRC review was risk-informed, and considered DOE's projected radiological risks, the impact of uncertainties, the potential impact of uncertainties that DOE did not consider, and the combined impact of uncertainties. The difference between the dose result of DOE's base case assessment and the performance criteria is large (i.e., the base case is several orders of magnitude below the performance criteria, creating a large safety margin). However, for the technical reasons discussed in this report, NRC determined that the margin is not likely to be as large as DOE projected, but that DOE demonstrated that the criteria would be met for most waste sources.

Overall Results and Conclusions

The NRC staff concludes the following, in all WMA C tanks and ancillary equipment except the plugged pipelines:

- DOE has 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 has 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 has 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 NRC staff concludes the following for the plugged pipelines:

- As a result of not having characterization data, the uncertainty in the inventory of plugged pipelines is too large. DOE has not demonstrated that it meets the above criteria for the plugged pipelines. The NRC recommends that DOE characterize the plugged pipelines to determine the concentration of radionuclides and the amount of free liquids that are present.

In addition, the NRC staff notes that DOE has indicated it plans to complete the following activities, which are necessary to validate the assumptions DOE makes in the Draft WIR Evaluation for WMA C and the WMA C PA:

1) DOE indicated that they will complete waste retrieval and characterization of waste from the C-301 catch tank and the CR-244 Vault to verify the assumptions made in the Draft WIR Evaluation. The NRC staff agrees this characterization is necessary to more accurately understand the inventory of material remaining.
2) DOE intends to complete the final closure cover design and verify its performance, including an evaluation of erosion protection. The NRC staff agrees this closure cover

design is needed because the closure cover can hinder inadvertent human intrusion into the residual waste (e.g., by excavation) for hundreds of years after closure.

3) DOE plans to select the final grout formulation and verify that it can achieve the necessary performance (i.e., confirm that it will have no shrinkage, will not degrade significantly over the period of analyses, and verify that the grout will have the target effective diffusion coefficients and hydraulic conductivities for the field-scale materials at high water-to-cement ratios). The NRC staff agrees with this approach because the final grout formulation will play a key role in limiting water contact with the waste and in limiting the release of radioactivity to slow diffusional release.

The NRC staff's review results and recommendations discussed for Criteria A, B, and C in this TER are being provided to DOE for consideration only and are not intended to represent any 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, which will contain the final waste determination of whether residual waste can be managed as LLW. The Final WIR Evaluation for WMA C may be used by DOE in the future, with several other decisions on other residual sources of radiological and chemical hazards, to decide on a closure plan for WMA C.

1 INTRODUCTION

1.1 Background and Purpose

The following sections provide the background and the regulatory history for Waste Management Area C (WMA C) at the Hanford Site and describe the purpose of this Technical Evaluation Report (TER).

1.1.1 Background and Regulatory History

The Ronald W. Reagan National Defense Authorization Act for Fiscal Year 2005 (NDAA) allows the Secretary of Energy to determine whether radioactive waste resulting from the reprocessing of spent nuclear fuel is not high-level waste (HLW) and, if so, then it may be managed as low-level waste (LLW) and identified as Waste Incidental to Reprocessing (WIR).[2] NDAA Section 3116(a) includes the requirement that the U.S. Department of Energy (DOE) consult with the U.S. Nuclear Regulatory Commission (NRC) on the DOE non-HLW determinations in the NDAA-Covered States of Idaho (ID) and South Carolina (SC). Although the NDAA only addresses consultation and monitoring activities within the NDAA-Covered States of Idaho and South Carolina,[3] the NRC also conducts technical reviews for WIR at sites in non-NDAA covered states (e.g., States of Washington and New York), at DOE's request. For non-NDAA covered states, the criteria for DOE waste determinations are specified by DOE Order 435.1.[4]

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 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.

As described in the Interagency Agreement (IA) with the NRC for Hanford activities, the DOE has requested that the NRC provide technical advice and "consultation" regarding DOE WIR determinations for closure of the HLW storage tanks in WMA C, as well as other tasks, at the Hanford Site.[5] The NRC staff has defined "consultation" as performing an independent technical review so that the NRC can reach its own conclusions as to whether DOE's proposed waste management approach satisfies the NDAA criteria. The NRC staff's guidance for the consultation activities are documented in NUREG-1854, "NRC Staff Guidance for Activities Related to U.S. Department of Energy Waste Determinations" (NRC, 2007).

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[2] As stated in NUREG-1854, the NDAA was enacted in October 2004. DOE uses technical analyses that are documented in a "waste determination" to evaluate whether waste is "incidental", or alternatively, is HLW. The concept behind this incidental waste is that the residual radioactive contamination of the material, if properly controlled, is sufficiently low such that it does not represent a hazard to public health and safety, and thus, does not need to be disposed of as HLW in a geologic repository.

[3] https://www.nrc.gov/waste/incidental-waste/wir-ndaa.html

[4] https://www.directives.doe.gov/directives-documents/400-series/0435.1-BOrder

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

1.1.2 Purpose of the TER

The DOE requested, by letter dated June 4, 2018 (Agencywide Document Access and Management System (ADAMS) Accession No. ML18156A447), that the NRC conduct a consultative review of its "Draft Waste Incidental to Reprocessing Evaluation for Closure of Waste Management Area C at the Hanford Site," dated March 2018 (Draft WIR Evaluation for WMA C) ((DOE, 2018), ADAMS Accession No. ML18156A446). The Draft WIR Evaluation also includes the "Performance Assessment of Waste Management Area C, Hanford Site, Washington," dated September 2016 ((DOE, 2016), or WMA C PA) (ADAMS Accession No. ML18099A127) and other supporting documents. DOE supplemented its initial request with additional information submitted in letter dated October 22, 2019 (ADAMS Accession No. ML19305A296).

The purpose of the Draft WIR Evaluation for WMA C and its supporting documents is to show that DOE's actions or proposed actions for managing waste residuals in grouted tanks and ancillary structures will satisfy the criteria in DOE Manual 435.1-1, *Radioactive Waste Management Manual*. These criteria must be met to determine that WMA C WIR is not HLW and may be managed as LLW.

The NRC staff's independent review of the Draft WIR Evaluation for WMA C and the supporting WMA C PA was conducted in accordance with the IA between the DOE and the NRC. In the IA, the DOE requested NRC consultative 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)[6] over DOE Manual 435.1-1 Criterion A (i.e., the removal of key radionuclides). Additionally, the DOE requested consultation pertaining to the reasonable expectation of compliance with the performance objectives for a compliance period of 1,000 years.

The Draft WIR Evaluation for WMA C addresses the stabilized residuals that will remain in the WMA C waste tanks and ancillary structures at the time of WMA C closure. This Draft WIR Evaluation, and the associated NRC staff technical evaluation, do not address other facilities or systems, waste removed from the waste tanks and ancillary structures, or the contaminated soil and groundwater from any previous leaks or releases at WMA C.

This TER documents the NRC staff's review of the Draft WIR Evaluation (and the PA) for the closure of WMA C. This TER also documents the NRC staff's review of DOE's responses to NRC's 2009 Request for Additional Information (RAI) associated with the NRC staff's evaluation of the Tank C-106 report (RPP-20658, 2008) (see Appendix A). In 2004, DOE requested NRC to review the waste retrieval activities for Tank C-106 within WMA C. However, it was decided that a more holistic evaluation of WMA C was needed, and the NRC staff's review of Tank C-106 was put on hold as it was enveloped by the development and review of the Draft WIR Evaluation for WMA C (ADAMS Accession No. ML18074A207).

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[6] 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.

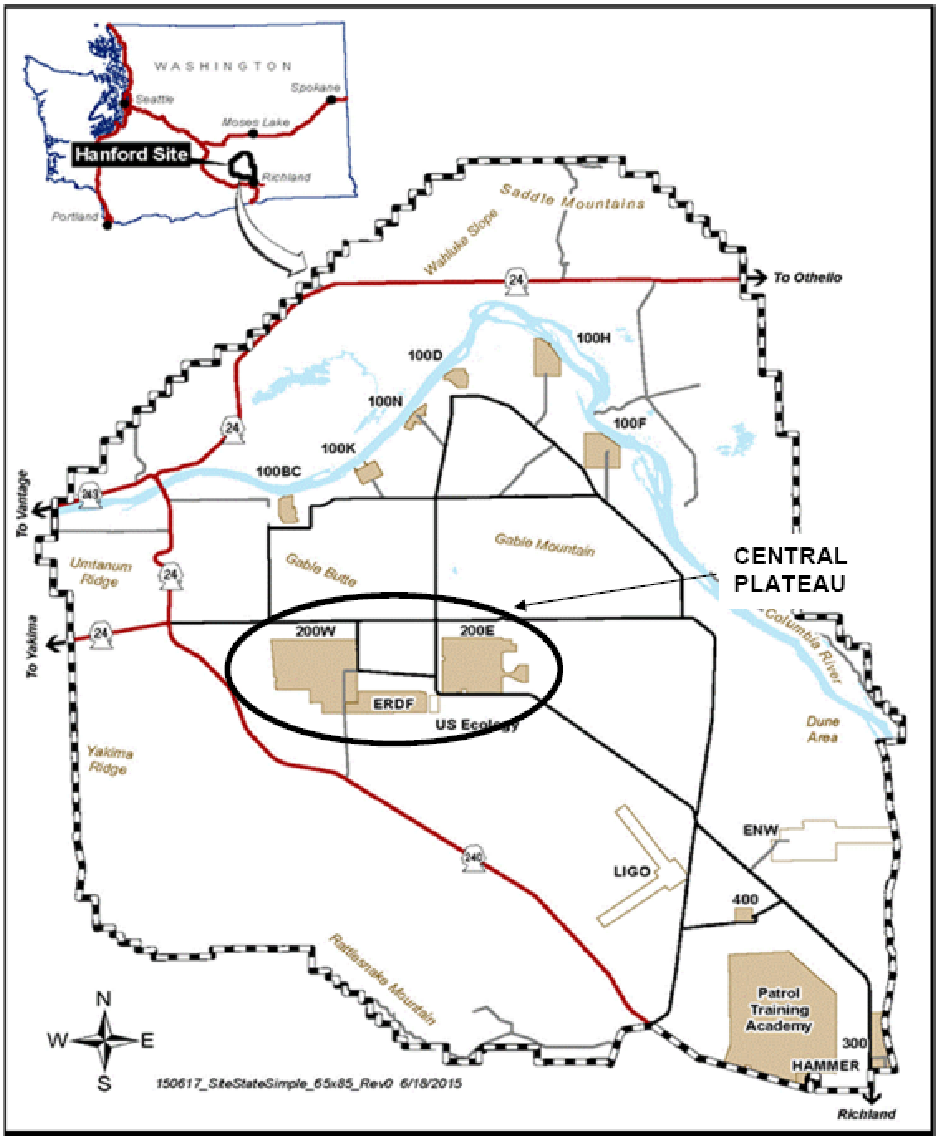

Figure 1-1 The Hanford Site
[Figure ES-1 in the WMA C PA (2016)]

1.2 Disposal Site Description

DOE's Hanford Site lies within the Columbia Plateau, a flood basalt plateau in the southeastern part of the State of Washington. The semi-arid Hanford Site is situated north of the city of Richland and at the confluence of the Yakima River with the Columbia River (see Figure 1-1). The 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 and provides an outer buffer for the nuclear materials storage, waste storage, and waste disposal areas within. About 6 percent of the of shrub-steppe and grasslands covered land has been disturbed and is actively used.

The WMA C site is in the east central portion of the 200 East area of the Hanford Site. The WMA C facility contains twelve 100-series tanks, four 200-series tanks, and a complex system of pipelines, diversion boxes, vaults, valve pits, and other miscellaneous structures.

1.2.1 Geography

The Columbia River runs through the Hanford Site. This area of the Columbia River Drainage Basin 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 scoured due to the massive floods. Since these floods, winds have locally reworked the flood sediments, loess (windblown silt), and the depositing dune sands in the lower elevations.

1.2.2 Meteorology and Climate

Climatological data for the Hanford Site is collected and processed at monitoring sites. Since the early 1980s, key information is 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.). 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 year 1995 held the wettest recorded year for the site, with 31.3 cm (12.3 in.) of precipitation (DOE, 2004).

The Cascade Mountain range in the western part of Washington influences the climate of the Hanford Site by means of its rain shadow effect. Summers are warm and dry, while winters are cool with occasional precipitation. This mountain range also affects the wind regime at the Hanford Site. Prevailing wind directions near the surface are 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.

1.2.3 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.

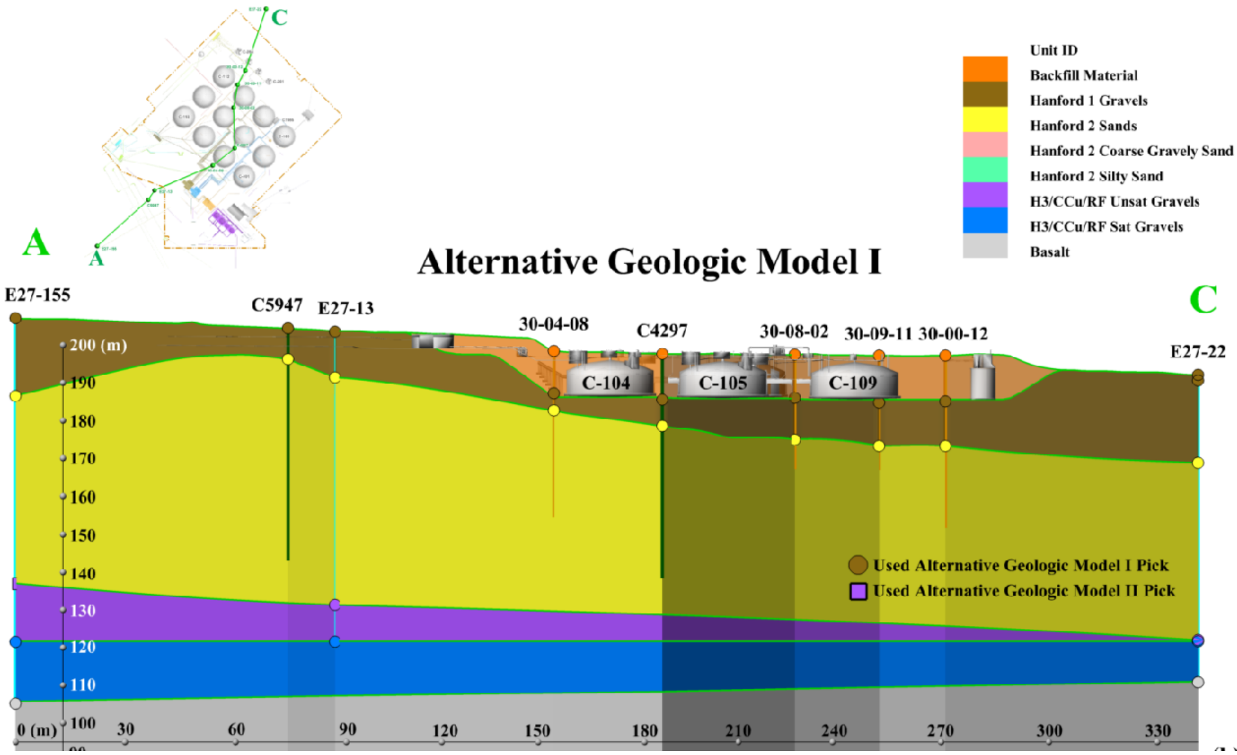

Figure 1-2 **Cross-Sectional Illustration of Hydrogeological Units in WMA C**
[Figure 3-38 in DOE, 2016]

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) thick or more, and the top of the basalt unit slopes gently to the southwest.

An undifferentiated Hanford H3 gravel, Cold Creek, and Ringold unit exists above the basalt unit. A paleochannel in the 200 East Area and near the WMA C 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 in this report as the H3/CCU/RF unit. Common thicknesses of the H3/CCU/RF under WMA C are 14 to 21 m (46 to 69 ft). See Figure 1-2.

The Hanford H2 sand formation lies above the H3/CCU/RF unit, and common thicknesses under the WMA C for the H2 unit are 45 to 50 m (148 to 164 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 unit thickens near WMA C and can reach thicknesses of 30 m (100 ft), but thicknesses between 9 m and 15 m (30 ft and 50 ft) are more common. Backfill exists at WMA C and other disturbed areas and is gravel-dominated; however, it can include cobbles, pebbles, and coarse to medium sand with some silt derived from the excavated H1 unit around tanks. Common thickness of the backfill under WMA C ranges from virtually nonexistent to over 20 m (66 ft).

Clastic dikes consist of multiple vertical layers of unconsolidated sand, silt, clay, and minor gravel. Clastic dikes have been documented in 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); however, no clastic dikes have been observed in WMA C. The deeper sections of clastic dikes appear to have many twists and turns and frequently will turn horizontally into a layer where they end.

1.2.4 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, 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. 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 under WMA C at the Hanford Site, approximately 70 m to 90 m (230 ft to 295 ft) and becomes even thicker as the water table drops in the Central Plateau area. Water was liberally used during operations for various purposes and frequently unable to drain away from the structures so that recharge rates to the aquifer were considerably higher for decades before the cleanup efforts began. 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. Natural recharge is estimated to be a few millimeters per year.

Much of the sediment in the unsaturated zone in the 200 Areas is contaminated due to the release or discharge of radioactive liquid waste by means of injection wells, French drains, cribs, ponds, and ditches. The report PNNL-SA-32152 (1999) estimated that 1.5 to 1.7 billion cubic meters (m^3) (396 to 449 billion gallons (gal)) of effluent were disposed of in the Hanford Site soils. Tritium (^{3}H), technetium-99 (^{99}Tc), and iodine-129 (^{129}I) are some of the more mobile radionuclides that can quickly move through the unsaturated zone, while cobolt-60 (^{60}Co), cesium-137 (^{137}Cs), and uranium (U) are some of the major contaminants that linger in the unsaturated sediments.

The saturated zone beneath the Hanford Site consists of the upper unconfined aquifer and the deeper basalt-confined aquifer. 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 down 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 and groundwater information indicates vertical communication 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 thicknesses in WMA C will be around 9 m to 12 m (30 ft to 40 ft). The unconfined aquifer at the Hanford Site is recharged in the elevated regions near the western boundary of the Hanford Site, and generally flows in an eastern and northern direction towards the Columbia River which is the primary discharge area for groundwater. The natural direction of flow beneath WMA C is toward the southeast; however, in the past it had been 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 WMA C (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 not low.

The groundwater quality in the Hanford area has been impacted by radiological and chemical contaminants resulting from past operations at the Hanford Site. 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. Section 1.4 in this report discusses current soil and aquifer contamination in more detail. DOE has a program that is addressing groundwater cleanup that is outside the scope of this review.

1.2.5 Demography, Natural Resources, and Land Use

Native Americans fished, hunted, and settled along the Columbia River and in the Hanford area for thousands of years. 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 percent increase in 10 years. The unincorporated population of Benton County was 33,227 in 2000.

The land use classification around the Hanford Site varies. At the Hanford Site, for example, cleanup of radioactive waste in facilities, soils, and groundwater is a major activity as well as radioactive material storage. Crushed rock, gravel, sand, and silt are currently the most commercially viable mineral resources since no deep natural gas has yet been successfully produced in the vicinity of the Hanford Site. 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 Columbia Basin 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.

1.3 Disposal Facility Description

Section 1.3 describes the 200 Areas in the Central Plateau of the Hanford Site and WMA C.

1.3.1 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, WA since 1943. Operations to make the raw materials for nuclear weapons for national defense continued until the late 1980s. In 1989, Hanford's mission shifted from production of weapons material to waste management and environmental cleanup under the care of the DOE. The cleanup of the site involves 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) and 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 Waste Treatment Plant is currently under construction within the 200 East Area. Waste recovered from the 200 Area tank farms will be pretreated and separated into HLW, that will be vitrified, and into low-activity waste (LAW) streams that will be vitrified or similarly immobilized. The 200 Areas also contains LLW disposal sites in addition to the 18 underground tank farms. In addition, land is leased by the State of Washington from the federal government and subleased to US Ecology, Inc. which operates a commercial LLW disposal facility occupying 40 hectares (100 acres) of this leased land just southwest of the 200 East Area.

1.3.2 Waste Management Area C

WMA C, or the 241-C Tank Farm, is roughly a few hundred meters wide and located in the 200 East Area. The site consists of 12 two million L (530,000 gal) 100-series single shelled tanks and four 210,000 L (55,000 gal) 200-series single shelled tanks. The 241-C tank farm is one of 18 tank farms at the Hanford Site. There have been 6.5 million liters (1.7 million gal) of waste that has been removed from the tanks at WMA C and transferred to other tanks. Drywells were installed around the 100-series single shelled tanks to aid in detecting waste release events. A complex waste transfer system of transfer lines, diversion boxes, vaults, valve pits, and other engineering structures or ancillary equipment were constructed to support the transfer and storage of waste between the tanks in WMA C.

1.3.2.1 Single-Shell Tanks

The 100-series tanks are 23 m (75 ft) in diameter, with a maximum operational height (cascade overflow level) of 4.9 m (16 ft) above the center of the dished tank base and another several meters to the base of the dome. There is a difference of 0.3 m (1 ft) between the lower center of the dished base and the base perimeter. The tanks are covered by a 0.38 m (1.25 ft) thick reinforced concrete domed top that is covered with several feet of backfill material (see Figure 1-3). The tanks were constructed with 0.95 cm (0.375 in) thick carbon steel lining the bottom and 0.64 cm (0.25 in) thick carbon steel lining the sides of a reinforced-concrete shell. The base of the tanks is up to 12 m (40 ft) below ground level. Tanks 241-C-101 (C-101) through 241-C-106 (C-106) have concrete pits. The other 100-series tanks are equipped with centrally located salt well pump pits. Tank pits are located on top of the tanks and provide access to the tanks, pumps, and associated monitoring equipment.

The four smaller 200-series tanks, 6 m (20 ft) in diameter and 7 m (24 ft) high, are piped to diversion box 241-C-252. The basic construction of the 200-series tanks is similar to the 100-series tanks. The 200-series tanks also have a base concrete slab over which a grout layer and a steel liner is present. Although the combined base concrete slab and grout layer thickness at the bottom of the four tanks is 0.178 m (7 in), the combined thickness of the 100-series tanks of 0.2 m (8 in) is applied to all tanks to simplifying diffusive transport calculations.

1.3.2.2 Ancillary Equipment

A waste transfer system of waste transfer lines or pipelines, seven diversion and valve boxes, four (244-CR) vault tanks, one (C-301) catch tank, valve pits, and other miscellaneous structures was constructed to support the transfer and storage of waste within the tanks of WMA C. Collectively, these are referred to as “ancillary equipment”. Except for the pipelines, valve pits, and other smaller miscellaneous structures, the remaining ancillary equipment will be filled with a cement-based free flowing grout, likely with the same grout formulation used in the tanks, that will harden with the intention of stabilizing the residual waste and providing structural stability.

The network of waste transfer lines or pipelines is especially complex. Multiple levels of piping were installed over time in WMA C and it is estimated that there are ~11 km (~7 mi) of waste transfer piping in WMA C. Estimated residual volumes in the pipes after closure is ~6,000 L (1,600 gal). Many of the pipelines are placed in encasements to provide secondary containment and protection of the pipeline. Pipelines from the diversion boxes to some of the tanks are supported by concrete viaducts. At ~3 m (9.8 ft) from the tank wall, the viaduct surface steps down and the void space between the pipes and the viaduct surface is grouted. The viaduct fans out from 0.8 m (2.6 ft) wide to 2.2 m (7.2 ft) wide to support the spread placement of the fill lines through the tank wall. In addition, the WMA C tanks are connected in four three-tank cascade series since each successive tank is situated 0.3 m (1 ft) lower than the feed tank, enabling fluid waste to flow through cascade lines from one tank to another as they were filled.

Figure 1-3 WMA C: Buried Single-Shelled Tanks and Ancillary Equipment

The 244-CR Vault is located south of the single shelled tanks. The 244-CR Vault is a two-level, multi-cell, reinforced-concrete structure constructed below grade, which contains two underground tanks with an estimated capacity of 170,343 L (45,000 gal) and two tanks with an estimated capacity of 55,494 L (14,700 gal). The routing of liquid waste to WMA C was accomplished using underground transfer lines, diversion boxes, and valve pits. The underground reinforced-concrete diversion boxes with pipeline connectors routed waste from one pipeline to another and were designed to contain any waste that leaked from HLW-transfer line connections. Leaked waste drained by gravity to nearby catch tanks. For example, Catch Tank C-301 was used to catch waste from the diversion boxes. The 244-CR Vault and associated diversion boxes 241-CR-151, 241-CR-152, and 241-CR-153 ceased operation in 1988 and DOE stated that roughly 98 percent of the liquid volume had been removed.

1.4 Current Soil Contamination at Waste Management Area C

Moisture movement through the unsaturated zone is the dominant process by which most contaminants in the unsaturated zone are transported. Past planned and unplanned releases of radioactive and hazardous wastes into the Hanford Site sediments are present-day potential sources of contamination. Contaminant plumes will generally expand as they are transported through the unsaturated zone and eventually reach the saturated zone. Contaminants may continue to move slowly downward for long periods (up to hundreds of years depending on recharge rates) after termination of liquid waste disposal or leaks.

1.4.1 Previous Releases of Radioactive Material and Estimated Inventory in the Soil

Intentional liquid-waste disposals, unplanned leaks, solid waste disposal, and leaks from underground tanks, including potential overflows from spare inlets and cascade lines, as well as transfer line leaks, have occurred in the 200 Area during historical operations. Numerous investigations have been carried out to identify the constituents and quantify the volumes. Past investigations include geophysical logging of monitoring wells in the unsaturated zone (drywells), geophysical logging and sampling of soils and groundwater from direct-push boreholes and wells, geophysical methods to obtain soil conductivity values, and evaluating past tank operations and waste processing information (RPP-RPT-42294, 2016). Key assessment parameters for estimating WMA C soil contamination inventory included: time and volume of the release, waste type and composition, and the mass of contaminants released from tanks or unplanned release sites.

When available, sample data was used near the time of release to estimate leak compositions; however, the information available was limited. Some constituents were analyzed, and composition estimates were largely based on waste types, total and spectral gamma geophysical logging measurements, and the Soil Inventory Model (RPP-ENV-33418, Rev. 4, 2016) historical process waste estimates. Tank farm leak assessments were performed in 2006, 2011, and 2014 and the results are summarized in Table 1-1. Another inventory model used to estimate leak compositions, the Hanford Defined Waste Model, is a spreadsheet-based engineering estimate of the chemical and radionuclide contents of the Hanford single-shell and double-shell tanks based on (1) process reactor fuel irradiation records, (2) separation plant dissolver charging records, (3) separation plant and tank farm process flowsheets, and (4) tank farm waste receipt and transfer records (RPP-RPT-59197, 2016). Chemical and radionuclide constituents evaluated in the Hanford Defined Waste Model are included in Table 1-2 (RPP-RPT-59197, 2016).

The radionuclide ^{99}Tc is a radionuclide of concern due to its relative prevalence and high mobility (i.e., its low capacity to sorb on solid particles). Releases from Tank C-105 are suspected of being the main source of the current ^{99}Tc concentration levels observed in the monitoring wells (RPP-RPT-59197, 2016). Although it is not certain if Tank C-105 is truly the source of ^{99}Tc, modeling results do indicate that most of the ^{99}Tc observed in the monitoring wells located southwest of the farm originated from the sources inside the farm, and not from the unplanned releases (UPR) that occurred after the Tank C-105 releases and away from the tanks (RPP-RPT-59197, 2016).

Table 1-1 Waste Releases from and Around WMA C (RPP-ENV-33418, 2016)

Tank/UPR	Volume (gal)	^{60}Co (Ci)	^{137}Cs (Ci)	^{99}Tc (Ci)	Basis
241-C-101	<37,000	1.7	900	0.25	Estimated waste release from surface level measurements from spare inlet or near the inlet
241-C-104	28,000	1.3	80	0.03	Cascade line leak next to C-104 (spare inlet)
241-C-105	2,000 to 20,500	0.1 to 1,4	4,200 to 42,000	1.0 to 9.8	Cascade line leak and possible leak at the base of tank (based on drywell activity)
241-C-108	18,000	0.8	52	0.02	Assumed cascade pipeline release. Volume based on ^{60}Co and soil moisture measurements
241-C-110	<2,000	0.6	350	3.4	Waste loss as the result of tank overflow through spare inlet nozzle
241-C-111	0				Liquid level changes believed to be a result of evaporation
241-C-112	7,000	0.3	20	0.0075	Transfer line leak from 252-C diversion box to tank C-112
241-C-200s	0				Calculations show liquid level decreases could be the result of evaporation
Other 241-C SSTs	0				Many tanks were overfilled but DOE did not have evidence of tank failure and insufficient data to estimate releases.
UPR-200-E-81	36,000	11	340	0.1	36,000-gal line leak based on RHO-CD-673
UPR-200-E-82	2,600	0.2	5,400	1.3	PUREX waste line leak in 1969
UPR-200-E-86	17,000	0.4	11,500	2.7	^{137}Cs based on 1971 sample. Site investigation suggests the volume estimate may be high.
Surface Releases	1,000	0.05	<3.0	0.001	Assume 10 pCi/g concentration of ^{137}Cs for top 10 ft (3 m) of soil inside WMA C
Total	169,100	18	60,700	17.5	

To convert gallons to L multiply by 3.78; To convert Ci to Bq multiply by 3.7×10^{10}

Table 1-2 Constituents Evaluated in the Hanford Defined Waste Model (RPP-RPT-59197, 2016)

Chemicals			Radionuclides			
Na	Ag	Si	^{3}H	^{113m}Cd	^{228}Ra	^{237}Np
Al	Mn	F	^{14}C	^{125}Sb	^{227}Ac	^{238}Pu
Fe	Ca	Cl	^{59}Ni	^{126}Sn	^{231}Pa	^{239}Pu
Cr	K	CCl_4	^{63}Ni	^{129}I	^{229}Th	^{240}Pu
Bi	U-Total	Butanol	^{60}Co	^{134}Cs	^{232}Th	^{241}Pu
La	NO_3	TBP	^{79}Se	^{137}Cs	^{232}U	^{242}Pu
Hg	NO_2	NPH	^{90}Sr	^{137m}Ba	^{233}U	^{241}Am
Zr	CO_3	NH_3	^{90}Y	^{151}Sm	^{234}U	^{243}Am
Pb	PO_4	$Fe(CN)_6$	^{93}Zr	^{152}Eu	^{235}U	^{242}Cm
Ni	SO_4		^{93m}Nb	^{154}Eu	^{236}U	^{243}Cm
			^{99}Tc	^{155}Eu	^{238}U	^{244}Cm
			^{106}Ru	^{226}Ra		

1.4.2 Radionuclide Transport of Soil Contaminants

Due to the large, temporary increase in recharge to the unconfined aquifer during operations at WMA C, the water table rose in the 200 Area. As a result of the water table rise, the hydraulic gradient increased and the direction of the groundwater flow changed to the northwest. It is predicted that the hydraulic gradient and direction of groundwater flow will change back to the southeast in a counterclockwise rotation as the groundwater regime continues to revert to its original state (DOE, 2016). DOE numerical models have simulated flow and transport of ^{99}Tc in an effort approximate observed field data for the time of arrival of ^{99}Tc to groundwater and concentration levels of ^{99}Tc in groundwater (RPP-RPT-59197, 2016). Due to the uncertainty of past conditions and the lack of data, assumptions were made with regards to past magnitude and direction of groundwater flow. These assumptions included the local direction of flow and hydraulic gradient at WMA C at the time the releases reached the water table while another assumption included the timing and inventory of the past releases. The simulation results contribute to developing a possible conceptual model on the transient conditions of the groundwater regime that shows changing concentration levels observed in monitoring wells.

Additional transient flow results involving transport of sorbing and nonsorbing contaminants showed that a conceptual model representing a one-dimensional passing of a contaminant front was unlikely. Instead, the concentration in the wells changed abruptly due to changes in the direction and magnitude of the hydraulic gradient. The rotation of the gradient from a northwestern to a southern direction continually changes the orientation of the groundwater plumes relative to the wells, and as the water table returns to more natural conditions, the rotation of the gradient will continue to alter the primary movement of the plumes to a more southeasterly direction. Additional simulation results project that the current high concentrations of ^{99}Tc are expected to decline over the next several decades as the plume disperses in the aquifer below WMA C (RPP-RPT-59197, 2016). Past leaks from the WMA C were the biggest contributors to the past and current concentration levels of ^{99}Tc.

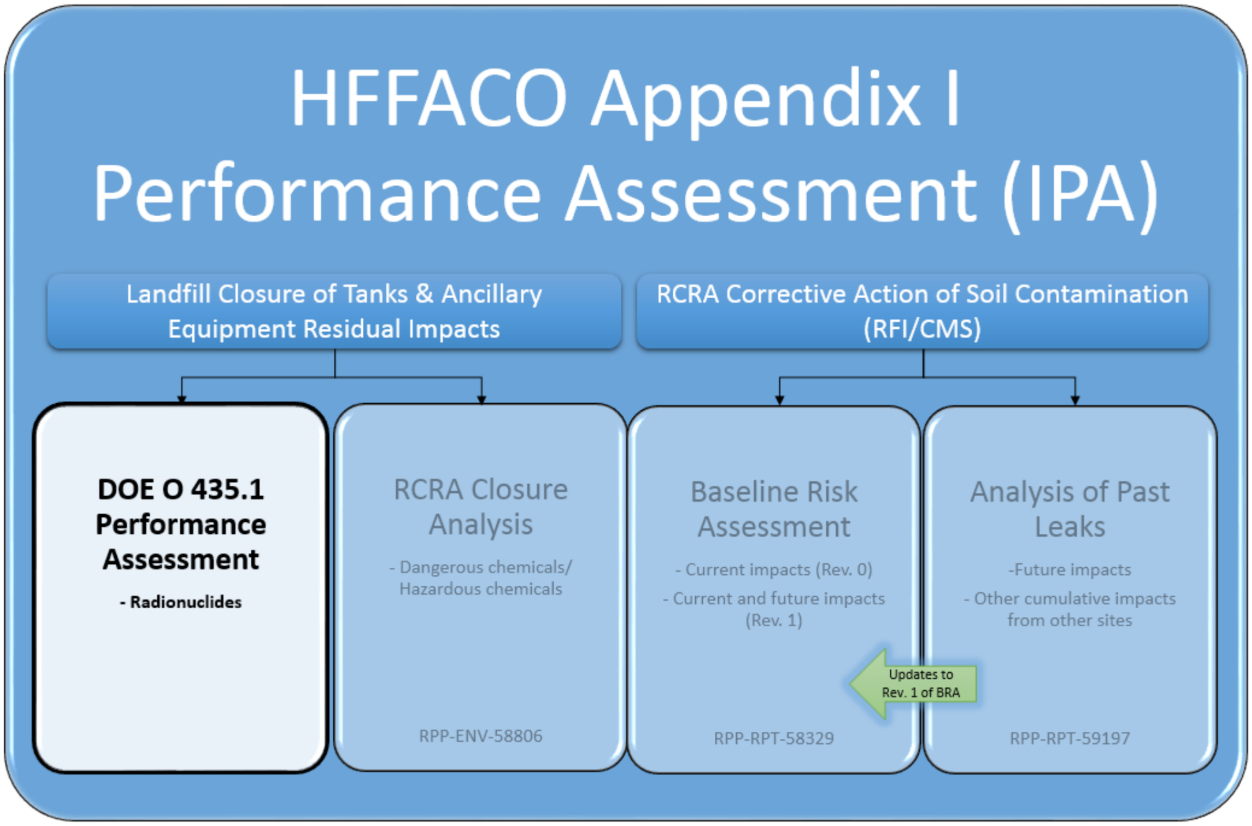

Figure 1-4 The Components of the Tri-Party Agreement's Appendix I Performance Assessment [Figure 1-2 in the WMA C PA (DOE, 2016)]

In comparison, WMA C past leaks or releases from residual wastes were not the most significant source for all other contaminants. For these contaminants, releases from upgradient sources were the more significant contributors to concentration levels. Other projection results showed no significant overlaps between future releases from residual wastes in a closed WMA C and the plumes from past leaks.

1.5 Waste Management Area C Closure Strategy

The Hanford Federal Facility Agreement and Consent Order or HFFACO (Tri-Party Agreement) was signed by DOE, Washington State Department of Ecology, and U. S. Environmental Protection Agency (EPA) in 1989. The Tri-Party Agreement is an enforceable agreement that requires DOE to clean up and dispose of radioactive and hazardous waste at the Hanford Site and close facilities that have been used to treat, store, or dispose of waste. Appendix I of the Tri-Party Agreement describes the single-shell tank system waste retrieval and closure process to be implemented. This action plan can be broken up into four components (see Figure 1-4):

1) A baseline risk assessment is completed at contaminated waste sites prior to remediation activities to establish a need for action. Guidance for conducting human health and ecological risk assessments for non-radiological and radiological contaminants in soils at WMA C, assuming no mitigating actions are taken, is based on the Resource Conservation and Recovery Act of 1976 (RCRA). [No NRC involvement]

2) An analysis of past leaks and an evaluation of future impacts to human and ecological receptors from both non-radiological and radiological contaminants in soils at the closed WMA C are completed. [No NRC involvement]

3) A RCRA Closure Analysis, or RCA, is completed that evaluates hazardous chemicals and waste residual contaminants in tanks and ancillary equipment at the closed WMA C. [No NRC involvement]

4) A PA is completed, which is a quantitative evaluation of potential releases into the environment and the resultant radiological doses, often performed using a computer model(s). In this case the PA is an evaluation of radioactive residual waste contaminants in tanks and ancillary equipment at the closed WMA C, based on DOE Order 435.1 (DOE, 2001b). The PA for WMA C should present a comprehensive, systematic analysis of the long-term impacts of a near-surface LLW facility after closure. The PA will also be used to support decisions related to WIR, that will be left at closure within tanks and ancillary equipment. The PA will be the primary tool used to demonstrate that the performance objectives of Subpart C of 10 CFR Part 61 will be met, and therefore, also the second criterion for the WIR evaluation process (see Criterion B below). It is also the focus of NRC staff's evaluation and the documentation in this TER. [NRC involvement]

The concept of WIR or non-HLW, is that some waste can be managed based on their risk to human health and the environment, rather than based on the origin of the waste. 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 shown not to require disposal in a geologic repository by means of a DOE analysis call a "waste determination". If it can be demonstrated that the waste does not pose the same amount of risk to human health and the environment as HLW and does not need to be disposed of as HLW, DOE may determine that the waste is non-HLW or WIR. DOE uses technical analyses documented in a waste determination to evaluate whether waste is incidental (i.e., WIR) or HLW. For the Hanford Site in the State of Washington, the criteria for DOE waste determinations are specified by DOE Order 435.1.

The WIR determination process, and NRC's staff evaluation of DOE's determination, is based on the criteria that is provided in DOE Order 435.1, and the related DOE Manual 435.1-1, which determines if WIR is not HLW and can be subsequently treated as either LLW or TRU waste. The criteria for determining if the waste can be managed as LLW (which is different from the criteria for managing it as TRU waste) 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 2012 Final Tank Closure and Waste Management Environmental Impact Statement (DOE, 2012) contained DOE's Record of Decision for closing WMA C, which requires a WIR determination of the tank residuals, a DOE Order 435.1 closure authorization/closure plan submittal, and RCRA closure plans (DOE, 2016). DOE's "Draft Waste Incidental to Reprocessing Evaluation for Closure of Waste Management Area C at the Hanford Site," (DOE, 2018), or Draft WIR Evaluation for WMA C (ADAMS Accession No. ML18156A446), together with the "Performance Assessment of Waste Management Area C, Hanford Site, Washington," (DOE, 2016), or WMA C PA (ADAMS Accession No. ML18099A127), provide a draft basis for a waste determination to be made at WMA C. The WIR determination and the decision to close the tanks will be made in accordance with DOE Order 435.1 and implemented through DOE Manual 435.1-1 (DOE, 2007). Closure of the individual single-shelled tanks and WMA C in its entirety will occur in three major steps: 1) retrieval of waste in the tanks, 2) filling the tanks with grout for stabilization, and 3) placement of an engineered surface cover barrier. The final state of a tank farm that is considered in the PA is a set of underground grouted tanks with associated ancillary equipment containing residual wastes that remain at the end of retrieval, covered by a modified RCRA Subtitle C surface cover.

DOE is consulting with the NRC and has stated it will consider the information in this TER and the comments from stakeholders before releasing the Final WIR Evaluation for WMA C containing the final waste determination of whether residual waste is WIR and can be managed as LLW.

1.6 NRC Review Approach

DOE has asked the NRC to provide technical advice and consultation regarding DOE WIR determinations for closure of the waste storage tanks in WMA C and other tasks in accordance with the IA. The NRC staff has defined "consultation" as performing an independent technical review so that the NRC can reach its own conclusions as to whether DOE's proposed waste management approach satisfies the criteria provided in DOE Manual 435.1-1, and that the waste can be managed as LLW. The NRC staff guidance for the consultation activities are documented in NUREG-1854, "NRC Staff Guidance for Activities Related to U.S. Department of Energy Waste Determinations" (NRC, 2007). At the conclusion of the NRC staff's review of each WIR determination, the NRC will provide DOE with a final TER documenting its findings.

DOE provided the Draft WIR Evaluation for WMA C (DOE, 2018) and the WMA C PA (DOE, 2016) to the NRC and requested that the NRC staff review the information as the basis for a draft waste determination for WMA C.

For the first criterion in DOE Manual 435.1-1 (Criterion A - removing key radionuclides to the maximum extent that is technically and economically practical), the NRC staff's review approach is that the purpose of the various criteria related to radionuclide removal is to minimize the inventory of highly radioactive radionuclides disposed of as incidental waste by removing the actual waste or by removing contaminated structures that contain waste. Frequently, this criterion can be satisfied by reducing the volume of residual waste in a contaminated structure (e.g., a tank, an evaporator) to the maximum extent practical. However, evaluating alternative methods of physically removing waste from a structure does not eliminate the need to consider (1) whether it would be practical to remove selected highly radioactive radionuclides from the waste (e.g., by chemical extraction) or (2) whether it would be practical to remove the contaminated structure for disposal instead of stabilizing it and disposing of it in place. Criterion A can involve the consideration of the risks and benefits of removing waste.

For the second criterion in DOE Manual 435.1-1 (Criterion B - will be managed to meet safety requirements comparable to the performance objectives set out in 10 CFR Part 61, Subpart C), 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. In addition, DOE requested that the NRC staff emphasize Criterion B over Criterion A in their review.

To evaluate compliance with the performance objective for the protection of the general population from releases of radioactivity (§61.41), the NRC staff review approach is to confirm that concentrations of radioactive material 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 performance objective for protection of individuals from inadvertent intrusion (§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. The performance objective for the protection of individuals during operations (§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 §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 (§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 §61.44 is limited to a review of site stability; however, because the stability of a disposal site is important to its long-term performance, the NRC staff reviewed whether the effects of site instabilities were adequately modeled or bounded in the PA and inadvertent intruder analysis.

For the third criterion in DOE Manual 435.1-1 (Criterion C - incorporating waste as 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), 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. Because the assessment of radionuclide concentrations is part of the assessment of radionuclide inventory and is essential to waste classification, the NRC staff review approach evaluates compliance with DOE Manual 435.1-1 requirements related to radionuclide concentrations and waste classification.

The decision to remediate the contaminated soil and groundwater underneath the tank farms will be made in accordance with DOE Manual 435.1-1. DOE is not consulting with the NRC concerning current soil contamination or the future decision to remediate the contaminated soil and groundwater underneath WMA C, and the NRC staff has not been involved in this component of the closure decision.

The NRC staff has carried out a risk-informed review of DOE's WMA C waste evaluation documents and information within the scope stipulated by DOE and NRC and documented the results in this TER. A risk-informed evaluation means that the review effort given to a technical topic during the evaluation be commensurate with the risk-significance of that topic; therefore, more attention and review time was given to features, processes, and events (FEPs) at WMA C that had the potential to significantly affect and influence 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. However, after the risk-informed review is complete, the rationale for the NRC staff's findings and recommendations are discussed and documented in this TER.

The NRC staff's review results and recommendations for the Draft WIR Evaluation for WMA C and supporting documents are provided for Criteria A, B, and C in Sections 2, 3, and 4, 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 area in the Draft WIR Evaluation for WMA C (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 provided in each subsection are collated into Table 5-1, which provides a listing of all the recommendations in this TER from Sections 2, 3, and 4. The recommendations are categorized as (1) applicable to the Draft WIR Evaluation for WMA C, (2) consider for future evaluations for other waste management areas, or (3) general technical recommendations that would generally improve the basis for the technical information but are not essential to the evaluation for WMA C. Recommendations that are categorized as "consider for future evaluations" could be risk-significant depending on the specific details of the waste management area being evaluated. The "general technical recommendations" are simply noted as best practices for performing waste evaluations.

2 CRITERION A – Key Radionuclides Removed

2.1 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.

2.1.1 Summary of 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 WMA C waste residuals, DOE included those radionuclides identified in the WMA C PA (DOE, 2016)[7] as important to satisfying 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.

The key radionuclides were developed from examining PA results for the groundwater pathway, the air pathway, the inadvertent intruder pathway, and from consideration of all pathways combined. The PA calculations had an initial list of 43 radionuclides considered.

For the groundwater pathway, DOE performed a screening analysis to look at the travel time of contaminants from the source (e.g., closed tank, ancillary equipment) to the water table. Only 7 of the 43 radionuclides were sufficiently mobile to reach the water table in 1,000 years (the DOE compliance period according to DOE Manual 435.1). These radionuclides included ^{60}Co, ^{3}H, ^{93m}Nb, ^{222}Rn, ^{99}Tc, ^{79}Se, and ^{129}I. An additional seven radionuclides arrived at the water table between 1,000 and 10,000 years. These radionuclides included ^{126}Sn, ^{14}C, ^{233}U, ^{234}U, ^{235}U, ^{236}U, and ^{238}U. Although the peak dose from the groundwater pathway was 4×10^{-6} mSv/yr (4×10^{-4} mrem/yr) and 1×10^{-3} mSv/yr (0.1 mrem/yr) during the 1,000 year and 10,000-year timeframes, respectively, DOE identified ^{99}Tc, ^{234}U, ^{238}U, and ^{129}I as key radionuclides. The air pathway analyses considered radionuclides that can partition into the gas phase from the dissolved phase. These radionuclides included ^{14}C, ^{3}H, ^{129}I, and ^{222}Rn. The peak air pathway dose was calculated to be 2×10^{-5} mSv/yr (2×10^{-3} mrem/yr). Only ^{3}H was a key radionuclide based on the air pathway results. DOE evaluated potential impacts to acute and chronic intruders. The key radionuclides, which comprised 95 percent of the intruder doses, were ^{90}Sr, ^{137}Cs, ^{239}Pu, ^{241}Am, and ^{240}Pu.

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[7] "Performance Assessment of Waste Management Area C, Hanford Site, Washington," (RPP-ENV-58782, 2016), (ML18099A131 and ML18099A136)

Table 2-1 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
^{3}H		X	X
^{14}C	X		
^{60}Co		X	
^{59}Ni	X		
^{63}Ni		X	
^{90}Sr		X	X
^{99}Tc	X		X
^{129}I	X		X
^{137}Cs		X	X
^{234}U			X
^{238}U			X
^{237}Np	X		
^{238}Pu	X		
^{239}Pu	X		X
^{240}Pu	X		
^{241}Pu	X		
^{242}Pu	X		
^{241}Am	X		
^{243}Am	X		
^{243}Cm	X		
^{244}Cm	X		

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 with two exceptions. Table 2 identifies ^{94}Nb and ^{242}Cm as radionuclides to consider when classifying waste. DOE indicated that those radionuclides do not exhibit significant activity in Hanford tank waste and, therefore, were not considered to be key radionuclides. Table 2-1 provides the radionuclides identified as key radionuclides by DOE in the Draft WIR Evaluation for WMA C.

2.1.2 NRC Evaluation of Identification of Key Radionuclides

DOE's approach to identifying key radionuclides is 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 residual waste at WMA C.

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 very small fractions of the dose limits. Though certain radionuclides (^{99}Tc for groundwater and ^{3}H for air) are the most significant for the pathway, if the radiological doses are very small and there is confidence in the magnitude of the doses, then an argument could be made that using a risk-informed approach, none of the radionuclides are significant for those pathways. The key to applying a risk-informed approach is to have proper quality assurance (QA) and model support to ensure the calculated values are correct and reasonable. 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 waste removal and remediation.

DOE also 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 uncertainty exposure scenarios should be included on a case-by-case basis. Some uncertainty exposure scenarios may represent a range of expected behaviors whereas others are purely speculative "what if" type of scenarios. It is appropriate for DOE to use subjective judgement to include additional key radionuclides based on the results of uncertainty exposure scenarios.

DOE indicated that key radionuclides with respect to worker safety were considered, however, no additional information was provided. Because the timeframe of potential worker exposures can be in the present and the pathways of exposure may be significantly different than the exposure pathways for a member of the public, it may be useful in future waste evaluations for DOE to provide a summary of past worker exposures and the radionuclides involved associated with tank waste remediation activities. Some short-lived radionuclides may be significant with respect to worker protection but may not be significant with respect to protection of members of the public.

DOE indicated that ^{242}Cm and ^{94}Nb were not included as part of the key radionuclides because these radionuclides do not exhibit significant activity in tank farm wastes. The NRC staff evaluated the inventories of these isotopes and how they were 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. This was because NRC did not anticipate commercial disposal (without reprocessing) of ^{94}Nb that would not be in activated metals. The limits for isotopes that are not in a metal form were set a factor of 10 less than isotopes that were in activated metals. Historically, ^{94}Nb was not included as part of DOE's Best Basis Inventory (BBI), not because it was determined not to be present in tank farm wastes (WHC-SD-WM-DP-025, 1992). The reason it was not included is because the main analytical laboratory did not have the capability to determine the concentrations of ^{94}Nb. To evaluate whether ^{242}Cm should be included as a key radionuclide, the NRC staff compared the tank farm inventory of ^{242}Cm, which was not included in the list of key radionuclides, with ^{243}Cm and ^{244}Cm, which were included. This comparison showed the inventory of ^{242}Cm is comparable to or greater than the other Cm isotopes (RPP-RPT-42323, Rev. 3, 2015). Though ^{242}Cm is relatively short-lived at 160 days, it has an important daughter product in ^{238}Pu. For completeness, it is recommended that ^{242}Cm and ^{94}Nb should be included as key radionuclides (Recommendation #1. See Table 5-1).

The NRC staff 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 would need to be revisited if the results of the technical analysis were to change materially.

Summary of Review

- The NRC staff evaluated DOE's approach to identifying key radionuclides and conclude that the approach is reasonable.
- There were no significant uncertainties associated with identifying key radionuclides.
- Recommendation #1 associated with this section can found in Table 5-1 of this report.

2.2 Removal to the Maximum Extent Practical

The first criterion in DOE Manual 435.1-1, (Criterion A) is that wastes "have been processed, or will be processed, to remove key radionuclides to the maximum extent that is technically and economically practical." DOE performed removal of waste from tanks at WMA C using a variety of methods and technologies.

2.2.1 Summary of 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 when and why waste retrieval operations were terminated and the alternative technologies that were considered.

2.2.1.1 Waste Removal Process and Performance

The tanks in WMA C contained a variety of wastes from different sources generated over an approximately 40-year timeframe. The wastes contained different radionuclides and chemical constituents. The physical and chemical properties of different wastes can vary substantially. In addition, after the waste was transferred to a tank, it could react with other waste in the tank or be modified during storage such as by self-boiling from heat generation. DOE indicated they tailored the technological approaches used to retrieve waste to address the physical, chemical and radionuclide properties of the waste to be removed and enhance the retrieval performance. Initial waste retrieval for tanks at WMA C began in 1998 (RPP-20577, 2007). DOE's approach was premised on the concept of achieving the limits of a retrieval technology for a specific tank and then determining what other technologies could be used to achieve additional waste removal to remove waste and key radionuclides to the maximum extent technically practical.

The DOE process for selection of waste retrieval technologies for WMA C tanks was based generally upon (RPP-PLAN-40145, 2015) and included consideration of:

- Which technologies were available at the time retrievals were performed,
- Known or assumed soundness of the tank being retrieved,

- Available tank access,
- Impact of the technology on available double-shelled tank storage space, and
- Expected effectiveness of the technology given a specific waste type.

DOE attempted to remove waste until the limit of technology waste was reached. DOE indicated the limit of technology was reached when the amount of waste removed during an operating period approached zero, and it was apparent that it was no longer technically practical to continue. DOE Manual 435.1-1 does not set numerical criteria associated with waste or key radionuclide removal to the maximum extent that is technically and economically practical. However, the HFFACO and Consent Decrees employ a 10.2 m^3 (360 ft^3) retrieval requirement or residual goal. The HFFACO language specifies that residual waste volumes in 100-series tanks will not exceed 10.2 m^3 (360 ft^3) or the limit of waste retrieval technology, whereas for 200-series tanks, the residual waste volumes will not exceed 0.85 m^3 (30 ft^3) or the limit of waste retrieval technology. DOE considered three types of information to demonstrate that the limit of a retrieval technology was reached:

1. In-tank photos and videos to observe and record the characteristics, form, and surface contours of waste,
2. Retrieval performance efficiency based on daily material balance calculations, and
3. Retrieval performance data trends to demonstrate that a consistent pattern was present and indicating that as much waste had been removed as practical.

The waste retrieval approaches and technologies DOE used to remove key radionuclides from tanks varied from tank to tank. This variation was driven by DOE's insight gained over time and because the chemical and physical characteristics of the waste can vary substantially between tanks.

The different forms of waste in the tanks included liquids, solids, and sludges. Some of the solids and sludges were light and moveable whereas others were very dense and strongly adhered to tank structures. Table 2-2 provides an overview of the different technologies DOE employed for the 100-series tanks. These technologies could be generally classified as sluicing/pumping, chemical, robotic, and vacuum. Usually multiple technologies were employed either in series or in parallel. The most heavily relied upon technology was some form of sluicing and pumping, with numerous variants. Robotic retrieval technology was used on two tanks (C-109, C-110) whereas vacuum retrieval was used on two 100-series tanks (C-105, C-107) and all four 200-series tanks. Figure 2-1 is a photograph taken for Tank C-108 used to monitor the progress and effectiveness of waste retrieval from the tank. Using photographs and videos of waste within the tanks allowed DOE to supplement the analytical data associated with bulk waste removal.

Table 2-2 provides the total initial volume of waste and the final volume of waste in each tank (or system). DOE indicated that 96 percent of the initial volume of waste in WMA C was removed in total. DOE also stated that the amount of radioactivity removed was 96 percent of the initial total radioactivity.

Table 2-2 Waste Removal Technologies Employed and Performance Result

Component Identification	Pumping and Sluicing				Robotic	Chemical		Vacuum			Initial Waste (gal)	Final Waste (gal)	% Removed
	Sluicing	MS	ERSS	HP	FoldTrack	Acid	Caustic	MARS-V	MARS-S	VRS			
C101		X	X	X							77500	4995	94
C102		X	X	X							316000	20200	94
C103		X									77800	2531	97
C104		X					X				259000	1600	99
C105			X				X	X			131700	4800	96
C106	X	X				X					230000	2770	99
C107		X							X		247000	10400[A]	96
C108		X					X				66000	2970[B]	96
C109		X			X		X				63400	1720	97
C110		X		X	X						178100	1773	99
C111		X	X	X			X				32500	4890	85
C112	X		X				X				104000	10100	90
C201										X	860	144	83
C202										X	1400	147	90
C203										X	2640	138	95
C204										X	1489	137	91
244-CR Vault											10726	1073	90[C]
C301											10500	1050	90[C]
Diversion boxes/pits											94	94	0
Pipelines											1600	1600	0

[A] This value reported by DOE is the 95 UCL and not the best estimate

[B] The referenced report provided a value of 4937 gal (18.7 m^3), which was revised in other reports to 2968 gal (11.2 m^3)

[C] Assumed future removal

To convert gallons to m^3, multiply by 0.003786

Figure 2-1 Image from Tank C-108 Video Used to Monitor Waste Retrieval

DOE's strategy was to achieve as much bulk waste removal as possible without targeting specific radionuclides. The removal efficiencies for individual tanks ranged from a low of 83% to a high of 99% (as NRC staff calculated in Table 2-2).

For ancillary equipment, DOE did not perform additional waste removal beyond normal operational flushing of lines and diversion boxes. DOE indicated that waste removal had not yet been completed for the CR-244 vault and catch Tank C-301. However, DOE completed assessments to develop inventory estimates for these systems (RPP-RPT-42323, Rev 3, 2015). DOE assumed in their assessments that technologies could be deployed which would result in an overall reduction of 90 percent of the waste volume from these two systems.

Waste Removal Process and Performance for Tank C-106

DOE used two retrieval technologies to remove waste from Tank C-106, sluicing followed by modified sluicing with acid dissolution. The initial waste volume was approximately 870 m^3 (230,000 gal) of which approximately 86 percent was sludge. The sluicing campaign combined with the liquid evaporation was successful in reducing the waste volume to 136 m^3 (35,986 gal).

Laboratory testing of oxalic acid dissolution of Tank C-106 waste demonstrated that approximately 70 percent of the solids would dissolve in oxalic acid (RPP-17158, 2003). Oxalic acid was added in discrete batches, circulated, allowed time to react, then removed. Table 2-3 provides the results from four acid dissolution batches. The waste retrieval efficiency dropped from 8% to 0.3% over the four-batch sequence. The tank had a single sluicer nozzle that, after the fourth acid dissolution batch, was determined to no longer be effective at reaching and mobilizing the remaining waste. The waste was piled at the furthest points from the nozzle.

Table 2-3 Performance of Modified Sluicing Batches for Tank C-106

Operation #	Volume Water Added (gal)	Volume Transferred (gal)	Volume Retrieved (gal)	Retrieval Efficiency (vol %)
1	56,160	61,033	4,873	8
2	46,472	48,079	1,607	3.3
3	59,228	60,085	857	1.4
4	83,501	83,718	217	0.3

To convert from gallons to m^3, multiply by 0.0283.

In response, DOE removed a mixer-educator pump and installed a second sluicing nozzle, which was effective in breaking up the remaining waste piles and moving waste to the pump. DOE determined that these methods had reached the limits of their technology and would not meet the HFFACO criteria (RPP-20658, 2008). The residual waste volume was estimated as 10.5 m^3 (370 ft^3) with a range of 7.8 m^3 to 13.2 m^3 (275 ft^3 to 467 ft^3).

2.2.1.2 Termination of Waste Removal Operations

DOE tracked operating data for waste retrieval to determine when a retrieval technology was no longer effective. Most retrieval campaigns pumped more than 4,000 m^3 (1 million gal) of slurry and some exceeded 20,000 m^3 (5 million gal). In many cases, charts and tables were provided containing the data used to determine when a technology was no longer effective.[8] For example, in modified sluicing operations for Tank C-106, the first operation had an estimated retrieval efficiency of 8% whereas by the fourth operation the retrieval efficiency had dropped to 0.3 %. Figure 2-2 provides a chart of the latter stages of the retrieval campaign for Tank C-101. The figure shows the decreasing efficiency of solids removal (cumulative) with continued pumping of slurry. The information provided in the Draft WIR Evaluation for WMA C was obtained from numerous waste retrieval reports that provided more detailed information (e.g., RPP-RPT-56796, 2014).

2.2.1.3 Alternative Treatment Technologies

DOE employed multiple technologies on tanks for which the removal goals were not met. Usually, the first technology was some variant of pumping and sluicing. Depending on the outcome of the first technology and the characteristics of the waste, DOE applied a second and possibly a third technology. Each technology was applied until DOE believed the limit of that technology had been met. Alternate treatment technologies were something other than traditional sluicing.

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[8] Appendix C of the 2010 Consent Decree states: The "limits of technology" means that the recovery rate of that retrieval technology for that tank is, or has become, limited to such an extent that it extends the retrieval duration to the point at which continued operation of the retrieval technology is not practicable, with the consideration of practicability to include matters such as risk reduction, facilitating tank closures, costs, the potential for exacerbating leaks, worker safety, and the overall impact on the tank waste retrieval and treatment mission. If 360 cubic feet (10.2 m^3) is reached with the first retrieval technology, the first retrieval technology shall be used to the "limits of technology" and a second retrieval technology shall not be required. If the waste residual goal of 360 ft^3 (10.2 m^3) is not achieved using the established two technologies, an additional retrieval technology established in a revised Tank Waste Remediation Work Plan shall be deployed to the limits of technology.

The following section provides an example of DOE's approach to considering alternative treatments. DOE report RPP-52290, Revision 1 (2012) provides the results of a practicality evaluation request to forego a third retrieval technology for Tank C-108. The tank initially contained approximately 250 m^3 (66,000 gal) of waste of which approximately 208 m^3 (55,000 gal) was sludge and the remainder was pumpable liquids. During its service life, the tank was used to store wastes including Hot Semiworks Plant waste, Plutonium Uranium Extraction Plant (PUREX) cladding waste, in-farm ferrocyanide scavenging waste, tributyl phosphate process waste, and bismuth phosphate first-cycle waste.

Modified sluicing was the first technology employed followed by caustic cleaning. The tank contained two sluicers and one slurry pump. The waste remaining after modified sluicing was estimated to be 26 m^3 (6,800 gal). Figure 2-3 shows the distribution of waste following modified sluicing. Most of the waste near the access points had been removed or moved. After modified sluicing, caustic dissolution was applied. Before beginning the caustic dissolution process, DOE determined that one of the sluicers had become plugged and they were unable to unplug it. DOE elected to proceed using one sluicer. Caustic dissolution is a multi-step process designed to target the mineral phases present in the tank. The extent of reaction at different steps was estimated by sampling the slurry in the tank and measuring the relevant chemical species (e.g., F^-, OH^-). The caustic dissolution process was estimated to remove 1,900 gal (7.2 m^3) of waste which would have resulted in 18.8 m^3 (5,000 gal). However, DOE revised the method of calculating residual volume and at the end of the caustic dissolution process the tank was estimated to contain 2,968 gal (11.2 m^3) (RPP-RPT-54757, 2014). Figure 2-4 shows the distribution of waste following caustic dissolution. The caustic dissolution process resulted in a cumulative worker dose of 0.8-person mSv (80-person mrem) between October and December 2011.

Because the volume remaining (11.2 m^3 [397 ft^3]) did not meet the waste residual goal of 10.2 m^3 (360 ft^3), DOE performed an evaluation to determine if a third retrieval technology should be deployed. The evaluation was limited to the technologies listed previously in Table 2-2. DOE examined the performance of the two technologies previously used and estimated that for most contaminants of concern (COC) the amount of mass/activity remaining was between 0.2% to 1.6% of the pre-retrieval inventory. Three exceptions noted were ^{90}Sr at 78%, uranium at 13%, and the actinides at 6%. Based on the expected performance (See Section 3) and resultant risk reduction, DOE concluded it was not practical to deploy a third technology on Tank C-108 (RPP-52290, 2012).

2.2.1.4 Cost-Benefit Analysis

For the 100-series tanks, the waste removal goal was achieved for 33 percent of the tanks. For the 200-series tanks, the waste removal goal was achieved for all the tanks. The 200-series tanks were much smaller and residual waste was more easily accessed. DOE indicated that approximately $750 million has been spent on waste removal and management for WMA C (DOE, 2019). To achieve a further 90 percent removal of waste from the 100-series tanks DOE estimated, it would cost between $78 million to $240 million.

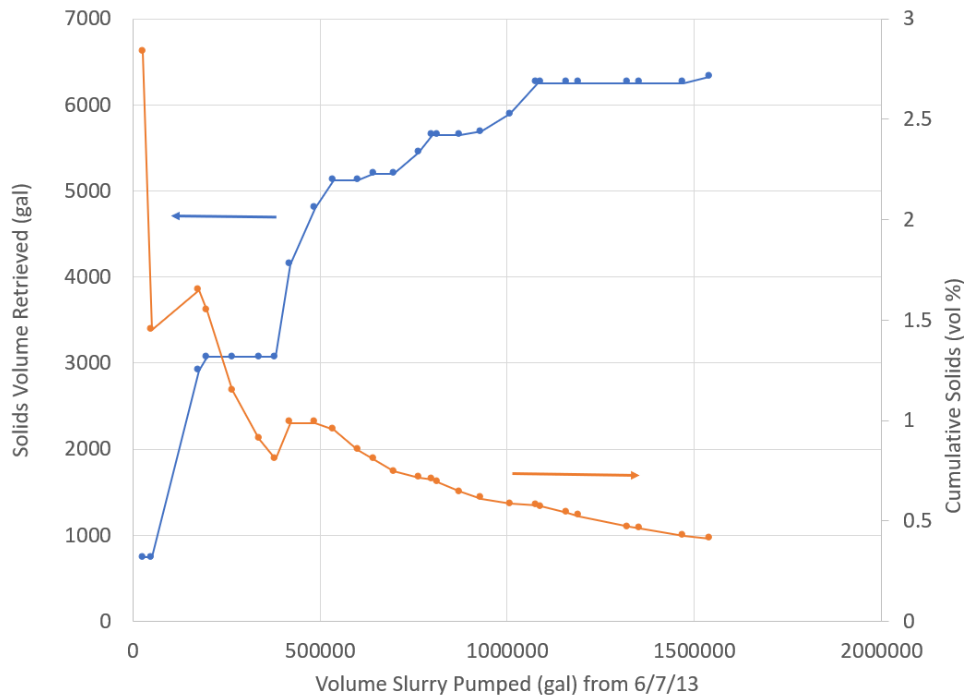
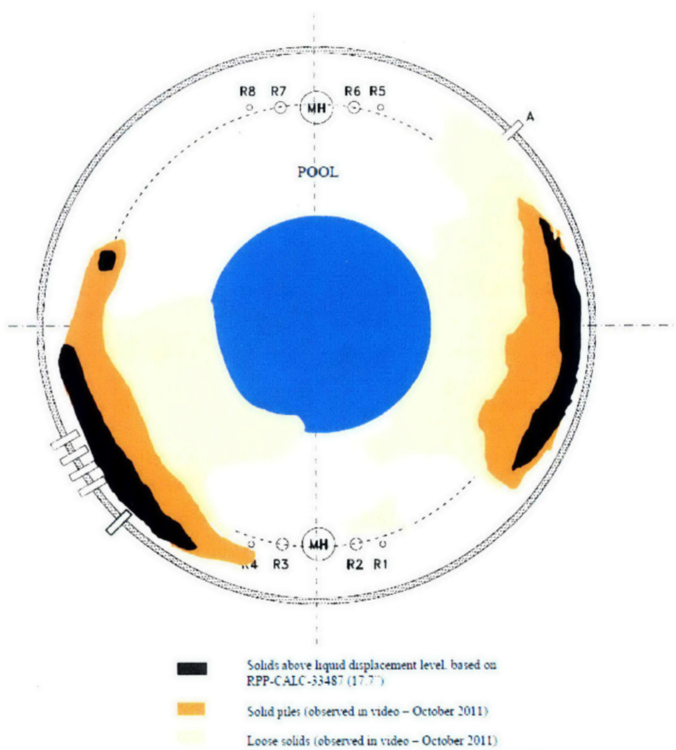


Figure 2-2 Cumulative Solids Removal for Tank C-101



Reference: RPP-CALC-33487, *Estimate of Waste Volume and Percent Retrieved for Tank 241-C-108.*

Figure 2-3 Distribution of Waste in Tank C-108 Following Modified Sluicing

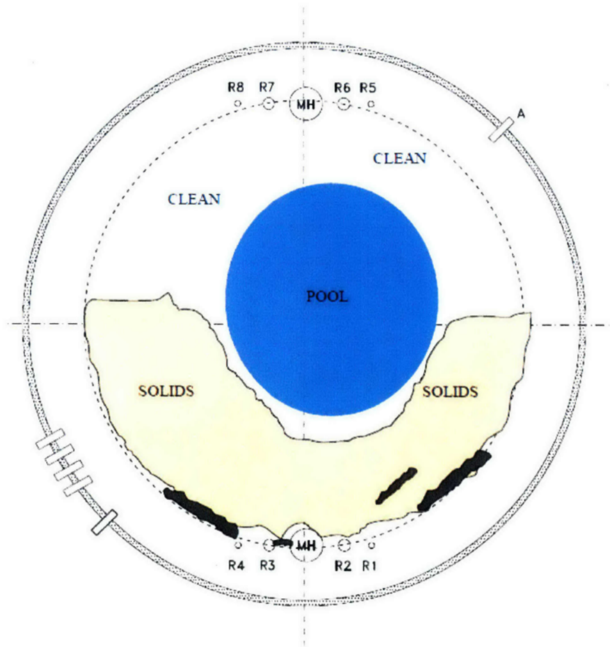


Figure 2-4 Distribution of Waste in Tank C-108 Following Caustic Dissolution

Based on the small projected impacts to the public (all-pathways groundwater dose of $4x10^{-6}$ mSv/yr ($4x10^{-4}$ mrem/yr), air pathway dose of $2x10^{-5}$ mSv/yr ($2x10^{-3}$ mrem/yr), increases in worker exposure 72 to 252 person mSv (7,200 to 25,200 person mrem), schedule delays (at least 6 years), and the acceptable results for potential impacts to the inadvertent intruder, DOE indicated additional removal of key radionuclides from tanks was not economically practical (DOE, 2018).

DOE indicated in their Draft WIR Evaluation for WMA C that diversion boxes and transfer pipelines were flushed as part of routine operations. DOE noted that cascade lines were gravity drained and at least one transfer line was known to be plugged. DOE believed that from a risk-informed perspective, additional removal of waste and key radionuclides from ancillary structures would not be practical. Because DOE's projected doses to the public and inadvertent intruders from ancillary structures were low, they indicated there would be negligible benefit from further waste removal and, therefore, DOE did not complete a formal cost-benefit analysis for ancillary structures. In response to NRC's RAI, DOE referenced the analysis of costs and benefits completed in report RPP-PLAN-47559, *Single-Shell Tank Waste Management Area C Pipeline Feasibility Evaluation* (2012). This report was a primary reference prepared as a scoping study to support WMA C closure planning that includes previous Hanford Site experience in accessing and characterizing contaminated structures as benchmarks for the analyses for WMA C. Section 7.3.1 of report RPP-PLAN-47559 provides a detailed examination of the costs, risks, and benefits of characterizing or removing waste transfer pipelines, and concludes that "further pipeline characterization or supplemental closure actions for protection of human health and the environment are not necessary when risks are balanced against the high cost and schedule impacts associated with these actions."

2.2.2 NRC Evaluation of Removal to Maximum Extent Practical

The NRC staff performed a risk-informed review of the information DOE provided in the Draft WIR Evaluation for WMA C, the WMA C PA, as well as numerous other supporting documents. The appropriateness of an approach to removal of key radionuclides to the maximum extent technically and economically practical is highly dependent on the projected impacts to the public from residual amounts of 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 bulk waste removal?
- In what ways does key radionuclide removal differ from bulk waste 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, with some iteration. However, removal of key radionuclides to the maximum extent technically and economically practical is a concept analogous to NRC's ALARA standard and is implemented using a risk-informed 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.

2.2.2.1 NRC Evaluation of Waste Removal Process

The NRC staff were able to verify DOE's bulk removal of 96 percent of the initial volume in the tanks. While this calculation is generally accurate, it includes some amount of initial water (i.e., supernate) added to the tanks prior to the initial operations to remove waste. In addition, removal of liquid waste from a tank is primarily determined by having a pump with an inlet at the lowest possible level in the tank. If the technical challenges of waste removal from tanks were limited to the pumping of liquid waste, then bulk removal percentages would be very high. It is the solid phases that can be difficult to remove. Liquid removal percentages are expected to be high and correspondingly the solid removal percentages would be lower.

Therefore, the bulk removal of 96 percent of the waste (overall average across all tanks) and approximately 96 percent of the activity should be considered in context. A bulk waste removal percentage of 96 percent is high considering the engineering challenges. However, the potential impacts associated with the residual waste (the 4 percent remaining) is the primary metric to consider, rather than the overall removal percentage. For example, high bulk removal percentages could correspond to high risk from residual waste and low bulk removal percentages could correspond to low risk – it depends on the composition of the waste and the system in which it remains. This aspect of the evaluation will be discussed in greater detail at the end of this section.

DOE used a structured approach of initially assessing the contents of the tanks and establishing a plan for waste removal for each tank. During implementation of these plans, the removal effectiveness was assessed, and some unexpected variances were encountered (e.g., equipment failure, waste did not react as expected) that required modifications to the approach. The NRC staff reviewed the retrieval plans and found them to have sufficient detail with respect to the technologies selected. Though the structured approach used by DOE may have advantages, it is likely to be more useful to preserve flexibility due to the uncertainties associated with deploying different technologies and the unique characteristics of different tank wastes.

Table 2-2 summarizes the technologies DOE used. Waste retrieval was accomplished with one or more retrieval technologies for each tank. Two or more different types of technologies (e.g., hydraulic pumping/sluicing, chemical, robotic, vacuum) were used for most of the tanks. Though DOE identified variants of pumping and sluicing as different technologies, they mainly differ only in the pressure and location of water or supernate supplied and in the circulation or recirculation rates. The variants may use different equipment; however, they all use equipment that relies on moving fluids around and are restricted by the fundamental fluid mechanics limitations of using liquids to mobilize solids of variable density and size. On the other hand, chemical, robotic, and vacuum-based techniques are fundamentally different. Waste for which the limits of hydraulic removal may be reached may be chemically altered or mechanically removed. The NRC staff believes that using different types of technologies is likely to result in the highest bulk removal rates.

DOE attempted to remove as much bulk waste as possible without targeting key radionuclides. DOE's premise was that if as much bulk waste was removed to the maximum extent technically and economically practical, then the key radionuclides would also be removed to the maximum extent technically and economically practical. The NRC staff generally agree with the objective of maximum bulk waste removal in the absence of deployable technologies to target removal of key radionuclides. Historically, Hanford Site staff do have experience with processing waste to remove isotopes (^{99}Tc, Pu-isotopes, ^{90}Sr, ^{137}Cs) (RPP-RPT-42323, Rev. 3, 2015), however, those wastes tended to be less confounded by the mixing of many different waste types as has occurred with the tank wastes.

There can be large differences between bulk waste removal and the removal of key radionuclides. As discussed previously in Section 2.2.1.3, removal of waste from Tank C-108 used modified sluicing followed by caustic dissolution. The bulk waste removal percentage was 96 percent, whereas the removal for the key radionuclide ^{90}Sr was only 22 percent. Almost 20 times more ^{90}Sr remained after using two different waste retrieval technologies than would have been anticipated by bulk waste removal. This result highlights that the ^{90}Sr was preferentially in the phases (solids) that were difficult to remove. Actinides and uranium isotopes were also preferentially distributed in the remaining solids. In the Draft WIR Evaluation for WMA C, DOE indicated that technological approaches were tailored considering the radionuclide properties of the waste, however, the NRC staff did not identify documentation to that effect.

DOE and other stakeholders set the volume removal goals or standards. As previously discussed, the HFFACO and Consent Decrees employ a retrieval requirement or residual goal. The HFFACO language specifies that residual waste volumes in 100-series tanks will not exceed 10.2 m^3 (360 ft^3) or the limit of waste retrieval technology, whereas for 200-series tanks, the residual waste volumes will not exceed 0.85 m^3 (30 ft^3) or the limit of waste retrieval technology. The volume-based standards used for WMA C tank residual retrieval are protective of public health and safety.

The NRC staff notes that not all waste volumes are equivalent from a risk perspective. Without considering the risk associated with those residuals, volume-based waste residual limits can result in less effort devoted to potentially riskier system components (e.g., plugged pipelines addressed in Section 3.10). Budgets and schedules are limiting, and if effort must be used on certain regulatory requirements, those resources may not be available for other potentially riskier systems. For example, considerable effort was undertaken to determine if retrieval activities for Tank C-106 achieved the 10.2 m^3 (360 ft^3) retrieval requirement. The best estimate of the volume remaining is 10.5 m^3 (370 ft^3), with an uncertainty range of 7.8 to 13.2 m^3 (275 to 467 ft^3). The impacts associated with this extra 0.28 m^3 (10 ft^3) that had not been retrieved can be estimated as:

1) The peak groundwater dose from all systems during the compliance period and sensitivity period is 4×10^{-6} mSv/yr (4×10^{-4} mrem/yr) and 1×10^{-3} mSv/yr (0.1 mrem/yr), respectively.
2) The peak groundwater concentration of ^{99}Tc from Tank C-106 is 18.5 Bq/m^3 (0.5 pCi/L) compared to 1,100 Bq/m^3 (30 pCi/L) overall. The proportion of Tank C-106 to the total is approximately 0.017.
3) The additional 0.28 m^3 (10 ft^3) represents a fraction of 0.028 of the residual waste volume requirement of 10.2 m^3 (360 ft^3).
4) The corresponding additional groundwater dose would be approximately 1.9×10^{-9} mSv (1.9×10^{-7} mrem) and 4.8×10^{-7} mSv (4.8×10^{-5} mrem) during the compliance and sensitivity periods, respectively.
5) The peak acute intruder dose for Tank C-106 was 0.0347 mSv (3.47 mrem). The corresponding additional acute intruder dose would be 1×10^{-3} mSv (0.1 mrem).

There are assumptions implicit with the calculations provided above. The calculations assume the DOE dose results are correct and free of error. The calculations assume the dose results are valid (e.g., proper input parameters and models were used). The calculations also assume the characterization of the residual waste and model estimates provides a reasonably accurate representation of the actual radionuclide concentrations in the waste. There are uncertainties and technical concerns with some aspects of DOE's results, those are discussed in detail in Section 3 of this report. However, given these assumptions, the impacts associated with the volume of waste above the retrieval requirement are quite low for Tank C-106.

There is considerable variance in the radiological composition of a unit volume of waste from system to system at the Hanford Site. In addition, the engineered and natural barriers that may be present to reduce the impacts to a member of the public from the waste may be considerably different from system to system. The NRC staff understands the purpose for the volume retrieval requirements or goals and believes they do provide a useful tool to direct initial waste removal activities and to evaluate progress. The NRC staff also acknowledges that NRC does not have a regulatory role at the Hanford Site and is only performing a consulting function to the

DOE on the Draft WIR Evaluation for WMA C. NRC's recommendation does not circumvent or have precedent over other agencies regulatory requirements or decisions. The volume-based retrieval standards may be more easily verified notwithstanding the uncertainties associated with the scale of the tanks and the limited access points; however, they may only reflect the radiological risk to a limited degree. It may be productive to revisit the volume-based retrieval standards for future waste retrieval activities at the Hanford Site. Use of a tiered approach to volume-based retrieval combined with risk insights may allow for reduced impacts to the public with fewer resources expended (Recommendation #2).

NRC Evaluation of Waste Removal Process for Tank C-106

Appendix A presents a complete discussion of DOE's waste retrieval campaigns for Tank C-106 at WMA C and the associated NRC staff reviews and RAIs issued in 2005 and 2009, along with DOE responses to the 2009 RAIs. Much of the information NRC requested in the 2009 RAIs was overtaken by changes to the DOE approach and updated information. For those items that were still relevant, the NRC staff incorporated them into the RAIs provided to the DOE on April 30, 2019 (NRC, 2019). A review of the waste removal process for Tank C-106 is presented here, see Appendix A for an evaluation of DOE's response to the NRC staff's 2009 RAIs.

DOE used two retrieval technologies to remove waste from Tank C-106 – sluicing and modified sluicing with acid dissolution. In addition, DOE made operational changes by adding an additional sluicing nozzle after observing the waste distribution in the system. DOE demonstrated that the removal efficiency (volume-based) decreased significantly with each subsequent modified sluicing batch. DOE determined that these methods had reached the limits of their technology and would not meet the HFFACO criteria (RPP-20658, 2008). The residual waste volume was estimated to be 10.5 m^3 (370 ft^3) with a range of 7.8 1o 13.2 m^3 (275 ft^3 to 467 ft^3).

The NRC staff believes the technologies used to remove waste from Tank C-106 were appropriate. The combined technologies were effective at removing approximately 99 percent of the waste from the tank. Though application of an additional modified sluicing batch or a third technology would likely remove the 0.28 m^3 (10 ft^3) necessary to achieve the HFFACO criteria, given the estimated impacts to members of the public and the uncertainty in the residual waste volume estimated, it would not appear to be technically or economically practical to remove more waste from Tank C-106. Greater emphasis should be placed on the uncertainty in the residual waste volume and waste composition estimates rather than the measures of central tendency of the volume remaining.

2.2.2.2 NRC Evaluation of Termination of Waste Removal Operations

DOE's approach to termination of waste removal operations was premised on the concept of achieving the limits of a retrieval technology for a specific tank and then determining what other technologies may be used to achieve additional waste removal, if needed. Next, DOE demonstrated that the limit of the additional technology was also reached. The NRC staff reviewed the information provided in the Draft WIR Evaluation for WMA C, tank waste retrieval plans, and tank waste retrieval reports.

For the following 100-series tanks (C-101, C-102, C-103, C-104, C-106, C-107, C-108, C-109, C-110, C-112), DOE provided charts and/or tabular information in the Draft WIR Evaluation for WMA C that the NRC staff reviewed to evaluate the basis for termination of waste removal operations. The NRC staff requested additional information about the information provided for select tanks, such as C-105 and C-111 (NRC, 2019). For retrieval that was based primarily on hydraulic technologies (sluicing and pumping), retrieval rates decrease at rates consistent with a stirred-tank reactor type of process. The analytical information shows the rates of waste retrieval decreased substantially. However, the rate of waste retrieval at termination does not appear to be well-linked to the potential impacts to public health and safety. Termination of waste retrieval at higher or lower rates of slurry retrieval could be appropriate depending on the amount of radioactivity in the waste and the associated long-term impacts to a member of the public. As discussed in Section 3.9, the impacts associated with the residuals in the 100- and 200- series tanks are estimated to be quite low, therefore, the NRC staff believes the termination of waste retrieval activities was appropriate.

For other retrieval technologies, the basis for terminating waste removal activities is more mixed. From a removal efficiency standpoint, DOE's rationale for terminating the chemical and mechanical processes was not always clear. For example:

- For Tank C-104, the extent of reaction with caustic recirculation time was approximately linear, suggesting that, with longer time, more reaction, and greater waste removal, could be achieved.
- For Tank C-108, the rate of increase of fluoride concentration in solution for the second water wash is comparable to the rate of increase for the first water wash (at comparable mixing times, adjusted for the different number of calendar days).
- For Tank C-109, the concentration of fluoride with circulation time shows continual increase (increasing effectiveness).
- For Tank C-110, the percent of waste retrieved was essentially linear with Mobile Retrieval Tool operating time (until a hydraulic leak occurred in the equipment).
- For Tank C-111, the percent of solids in the slurry increased with hot water additions, and the hydroxide concentration during continual recirculation decreased but had not reached a clear point of reaction completion.

Figure 2-5 shows the hydroxide concentration during caustic dissolution in various WMA C tanks. Some of the decrease is a result of water additions, and not due to reaction with the waste. For example, DOE indicated in RPP-RPT-59363 Rev 00A (2016) that 2,260 gal (8.6 m^3) of water was added to Tank C-111 during the caustic dissolution. Approximately half of the decrease in hydroxide concentrations are the result of the water additions.

Because the chemical techniques rely on chemical reactions, additional time would typically allow for a greater extent of the reaction to occur. Laboratory testing was completed prior to operations in some tanks. The NRC staff agrees with using laboratory testing to help design the retrieval program for each tank. However, in some cases, the laboratory tests and actual

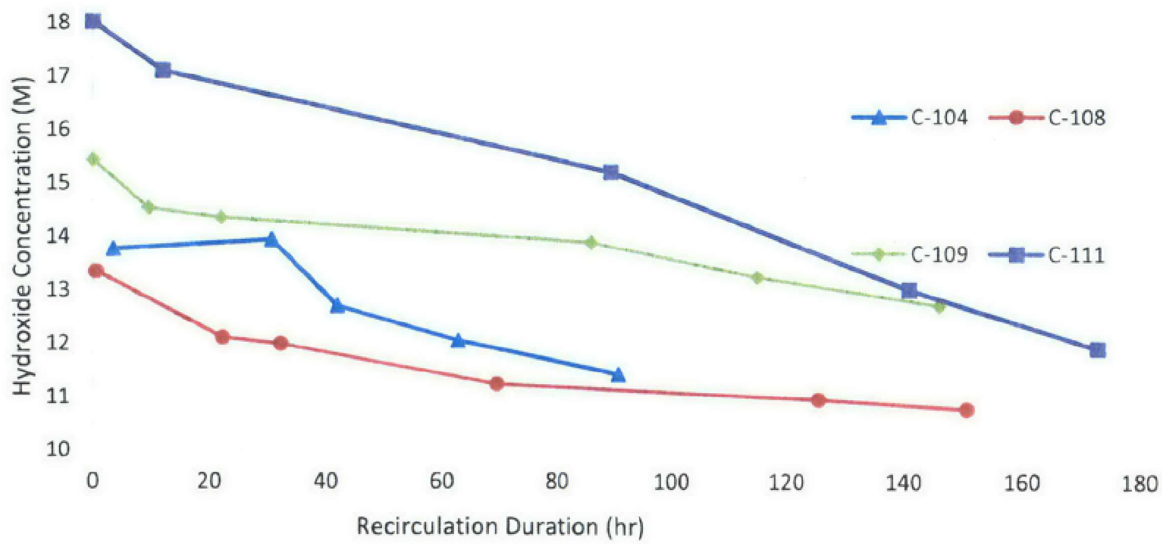


Figure 2-5 Hydroxide Concentration During Caustic Dissolution

experience in the tank deviated significantly. For example, the bulk waste in Tank C-111 was very difficult to remove with sluicing, whereas laboratory tests predicted that it would be able to be removed. Comparison of field observed results to the anticipated results based on laboratory testing may not always yield a robust basis for terminating retrieval activities, if the actual waste in the tank has different phases present or is in a different physical configuration and state such that the conditions for contacting reactants with the waste are different. If there is disagreement between field data and laboratory data, then the rate of change of the field data should be the primary input to the decision of when to terminate waste retrieval. The criteria for terminating retrieval of waste by chemical means should be adjusted based on field experience when field experience differs from laboratory experience (Recommendation #3).

2.2.2.3 NRC Evaluation of Alternative Treatment Technologies

In Section 2.3.3 of the Draft WIR Evaluation for WMA C, DOE provided a summary of the technologies that were deployed. However, it did not indicate if additional technologies or other operational changes were considered to increase waste removal. Operational changes were noted in data retrieval reports that the NRC staff reviewed and some of them were effective at increasing waste retrieval.

In RPP-RPT-44139 (2014), DOE provides a summary and roadmap of nuclear waste retrieval technology. The report summarizes tank waste retrieval developmental activities completed from 2010 to 2014. The report discusses a 2009 DOE workshop to evaluate waste retrieval technologies that included representatives from the commercial mining industry. The report indicates that waste retrieval system deployments have predominantly been limited to installation in existing risers, which has constrained the use of commercial-off-the-shelf hardware and required miniaturization of equipment beyond commercial experience.

Limitation of access points seems to play a key role in the amount of waste removed from the 100-series tanks, since waste removal is highly reliant on hydraulic-based processes. In some cases, DOE created new access locations (e.g., a large central riser was drilled into the center of the tank dome to install the Mobile Arm Retrieval System in Tank C-107) whereas in other cases they did not (e.g., Tank C-108).

Figures 2-3 and 2-4 show the importance of the number of sluicers in a tank. When two sluicers were available in Tank C-108, the waste closest to the sluicers was removed, resulting in piles of waste on the periphery of the tank at the farthest distance from the sluicers. The motive force that can be applied to the waste to direct it to the pump for removal decreases with increasing distance. After one of those sluicers was plugged, the waste was redistributed to the half of the tank farthest from the operational sluicer. The residual waste in the tank was able to be moved around; however, the waste could not be removed using the technology employed. Similarly, for Tank C-110, DOE indicated in the Draft WIR Evaluation for WMA C that the waste could be mobilized with the sluicers but would settle and could not be easily retrieved with the existing installed equipment.

At the early stage of hydraulic removal processes, the smallest particles and most soluble waste is mobilized. At the latter stages of hydraulic removal, only the largest, insoluble, and most dense particles remain, making it increasingly difficult to remove the waste with hydraulic means. In RPP-RPT-44139 (2014), DOE indicated that using three sluicers can result in a large percentage of the waste being removed from a tank. A primary consideration for waste retrieval should be to determine what access is available and, if necessary, increase the number, type, or location of access points prior to beginning waste retrieval (Recommendation #4).

DOE evaluated technologies for hard-to-remove waste, focusing on mechanical and chemical processes. Some of the mechanical technologies DOE considered included the In-Tank Vehicle, the Sand Mantis, the Salt Mantis, Large Remote Operated Vehicle, Sycamore Construction Retrieval System (mechanical dredge), and the FoldTrak Retrieval System. Some of the benefits of in tank vehicles include (RPP-RPT-44139, 2014):

- The use of plow blades that can push waste that is outside the area of influence of the sluicers into the area of influence of the sluicers or into the area influenced by the waste transfer pump.
- The weight of the vehicle driving over waste can reduce the particle size of the waste, making it more readily mobilized by the sluice stream.
- A particle size reduction end effector using rotating flairs or cutting blades could be mounted onto an in-tank vehicle or robotic arm.
- The in-tank vehicle can reach and mobilize waste that is difficult to mobilize with sluicers alone.

DOE successfully deployed the FoldTrak in Tank C-110 with essentially linear waste removal efficiency with operating time until a hydraulic leak occurred. The other mechanical technologies have not yet been deployed at the Hanford Site, although some technologies (e.g., Sand Mantis, Salt Mantis) have been deployed at the Savannah River Site.

DOE began tank waste retrieval in 1998 and completed retrieval in 2017. During this almost 20-year period, DOE gained considerable experience and invested heavily in the development of new technology to provide robust and efficient solutions to technical challenges. Technology evolves very rapidly such that new technology may become available that is not subject to the limitations of older technology. Technologies that have been deployed in the past have the benefit of experience to deploy them in the future, and newer technologies will have the risk of being unproven. It is recommended that DOE periodically review new technologies that may be deployed to increase waste retrieval (Recommendation #5). Rapid advances are being made in the field of robotics. The NRC staff recommends that DOE should evaluate more mechanical robotics solutions for the Hanford site's remaining tank farms (Recommendation #6). Mechanical solutions can be used to target the difficult to remove waste that can contain a high proportion of the key radionuclides. Considering the large amount of funds invested into removing waste from WMA C, the amount invested in technology development was not clear in the Draft WIR Evaluation for WMA C. In the Draft WIR Evaluation, DOE did not provide the cost of individual technologies to allow a better assessment of alternative treatment technologies.

For many tanks, approximately 75 percent of the waste was retrieved in the first approximately 25 percent of the volume of slurry pumped. The requirement to reach the limit of technology for the first technology used before moving to a second (or third technology) may result in an inefficient use of resources and ultimately higher risk (given budget and schedule constraints). The NRC staff recommends this approach be reconsidered (Recommendation #7). As previously mentioned, sluicing through limited access points is unable to overcome the fundamental fluid mechanics limitations of suspending large, dense particles over long distances. Earlier deployment of mechanical and chemical technologies reserved for hard-to-remove heels may result in faster rates of waste removal per total volume of slurry pumped. Because of the reduced amount of supernate used and time saved, other technologies targeting the specific physical and chemical characteristics of the remaining solids may be deployed. Given DOE's description of the characteristics of the hard-to-remove wastes, dredging (suction and mechanical) would appear to be a strong candidate technology to explore in greater detail. Use of a FoldTrak type vehicle, combined with a mechanical dredge, could efficiently remove solids without using large volumes of water or supernate.

2.2.2.4 NRC Evaluation of Cost-Benefit Analysis

DOE developed cost-benefit information for further waste removal from the 100-series tanks, as the waste removal goal was achieved for only one third of the tanks. DOE indicated that approximately $750 million has been spent on waste removal and management for WMA C (DOE, 2019). To achieve a further 90 percent removal of waste from the 100-series tanks, DOE estimated it would cost between $78 million to $240 million. The value of an additional 90 percent removal was selected for cost-benefit purposes and was not tied to particular removal goals. Based on the small projected impacts to the public (all-pathways groundwater dose of 4×10^{-6} mSv/yr (4×10^{-4} mrem/yr), air pathway dose of 2×10^{-5} mSv/yr (2×10^{-3} mrem/yr), increases in worker exposure 72 to 252 person-mSv (7,200 to 25,200 person mrem), schedule delays (at least 6 years), and the acceptable results for potential impacts to the inadvertent intruder, DOE indicated additional removal of key radionuclides from tanks was not economically practical.

Though the NRC staff has technical concerns with some aspects of DOE's projected dose impacts that could make the doses larger and more uncertain than what DOE projects (as discussed in Section 3), the doses from the 100-series tanks residuals are still well within the limits for the performance criteria. DOE's analysis of further waste removal is a "what if" type of analysis, which is appropriate to communicate relative impacts associated with potential future waste removal. As waste is removed from the tanks, it becomes increasing difficult to remove the waste that remains. With current technology, the NRC staff believes that DOE's 90 percent additional waste removal may be very difficult to achieve in practice, therefore, the estimated costs would be even higher than estimated by DOE.

Though it is useful to consider worker doses, worker doses are not an imposed risk; they are a risk accepted by the worker. Worker doses are not directly comparable to doses to a member of the public. In addition, the use of advanced technologies (e.g., robotics) can reduce the impacts to workers but would increase costs. There is a direct tradeoff between economic impacts and worker doses.

DOE indicated that at least 6 years of schedule delays would occur as a result of removal of an additional 90 percent of the residual waste. It is important to continue to make progress in retrieving and treating waste remaining in the tank farms at the Hanford Site. However, because of the high radiation fields, the need for contamination control, industrial hazards, and complex engineering tasks, not many actions associated with the WMA C tank farm can be accomplished quickly. The estimated schedule delays by DOE are consistent with previous experience in removing waste from the tanks. The tanks are being operated well beyond their design life. With increasing time, the likelihood of an unforeseen event (e.g., waste reaction, excessive corrosion, operational error, deterioration of access or equipment, seismic event) occurring increases. When the bulk of the waste has been removed from the tanks, the tanks are in a much lower risk state, because if something were to happen, there is less material in the tanks (~ 4% of the waste remains). A delay in final closure may allow the development of new technologies that could remove more waste from the tanks with significantly lower costs. However, in the case of the 100- and 200-series tanks at WMA C, a further delay in closure does not appear to be necessary because the projected impacts to a member of the public are quite low.

Because DOE's projected doses to the public and inadvertent intruders from ancillary structures were low, they indicated there would be negligible benefit from further waste removal, and therefore, did not complete a formal cost-benefit analysis for ancillary structures. The NRC staff finds that DOE did not provide adequate information to justify their conclusion. As discussed in Section 3.10, DOE evaluated the dose to an inadvertent intruder from a plugged pipeline. DOE estimated the chronic dose to an inadvertent intruder as 1.6 mSv/yr (160 mrem/yr) which exceeds the DOE performance objective of 1.0 mSv/yr (100 mrem/yr). This dose is driven by short-lived isotopes (e.g., ^{90}Sr and ^{137}Cs) and decreases roughly an order of magnitude from 100 to 200 years after closure.

Table 2-4 Potential Inadvertent Intruder Doses from Ancillary Equipment

Exposure Scenario Type	Ancillary (Avg. Waste)	Plugged Line (Avg. Waste)	Plugged Line (Transferred Waste)
Acute	36.0 mrem#	700 mrem*	7,000 mrem* +/-
Chronic	8.2 mrem	160 mrem	1,600 mrem* +/-

To calculate mSv, divide the mrem by 100
* Not calculated by DOE, estimated by NRC. The concentrations of waste in plugged pipelines is highly uncertain without characterization data for the plugged pipelines. Doses could be much higher or much lower.

DOE assumed the inventory in the plugged pipeline was the same as the average inventory in the residual waste in WMA C. The problem with this assumption is that the pipeline(s) plugged at a discrete point in time during early operations at the tank farm. The wastes being transferred at that time were, in some cases, very concentrated and could have had much higher concentrations of waste, potentially orders of magnitude higher, than the average present-day concentrations in WMA C, which have been subject to considerable mixing with other waste streams and process waters. In addition, in the WMA C PA, DOE did not provide the acute dose to an intruder from a plugged pipeline.

Table 2-4 summarizes DOE's dose results to the inadvertent intruder as well as the NRC staff's estimates. The acute dose into other ancillary structures was 0.36 mSv (36.0 mrem), whereas the chronic dose was 0.082 mSv (8.2 mrem), or a ratio of 4.4. Applying a similar ratio to the 1.6 mSv (160 mrem) chronic dose that DOE calculated would result in approximately 7 mSv (700 mrem) dose to the acute intruder from a plugged pipeline. The NRC staff show for comparison dose results if the waste in a plugged pipeline were more concentrated (e.g., 70 mSv (7,000 mrem). Solids generally have more radioactivity than liquids, and if plugging were the result of precipitation processes, the radioactivity in plugged lines could be much higher than the average waste. Likewise, if the waste that caused the plugging was less concentrated than the average waste, the doses would be correspondingly lower (e.g., 0.07 mSv (7 mrem)).

The concentration of waste in the plugged pipelines is unknown. Though other ancillary equipment may pose limited risk to a member of the public, a plugged pipeline could pose high risk depending on the concentrations present. DOE indicated that the plugged pipeline (and cascade lines) represent a small percentage over the overall length of piping that will remain in WMA C. While this statement is accurate, all waste that is disposed in the near-surface environment must be suitable for disposal irrespective of what other waste may be co-located with it. If the plugged pipeline were to be inadvertently intruded upon, the radiological dose the intruder would receive would not be changed by the fact that another intruder had previously intercepted a pipeline with more "average" concentrations. The NRC staff's review of development of the inventory for pipelines is discussed in Section 3.5.

Summary of Review

- The NRC staff reviewed DOE's information to demonstrate removal of key radionuclides to the maximum extent practical. DOE's approach to removal of key radionuclides by implementing bulk waste removal to the limit of technology is appropriate in the absence of deployable technologies to target removal of key radionuclides in the tanks. DOE demonstrated removal of key radionuclides to the maximum extent technically and economically practical for the 100-series tanks, 200-series tanks, and most of the ancillary equipment. Because of inventory uncertainties, DOE did not demonstrate removal of key radionuclides to the maximum extent practical from plugged pipelines.

- The inventory of key radionuclide in plugged pipelines is extremely uncertain and DOE has not removed any waste from the plugged pipelines.
- Recommendations #2 through #7 associated with this section can found in Table 5-1 of this report

2.3 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. The NRC staff evaluated the identification of key radionuclides and the removal of key radionuclides to the maximum extent technically and economically practical. The latter aspect included assessing waste removal processes, termination of waste removal, DOE's consideration of alternative treatment technologies, and cost-benefit analyses. The NRC staff performed a risk-informed review of the information provided. The acceptability of waste retrieval is strongly coupled to the technical acceptability of the PA and other analyses. Explicit assumptions associated with the NRC staff conclusions are provided below. In the NRC staff's professional judgment, these assumptions are likely to be essentially confirmed. In the event of significant deviations, the conclusions found below may no longer be valid. Unless explicitly stated, remaining uncertainties associated with demonstrating that other criterion will be met are not found to be significant with respect to the conclusions provided here.

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

- Ninety percent of the waste volume will be removed from catch tank C-301 and the 244-CR vault and associated components.
- The volume of waste in the pipelines will not materially differ from the five-volume percent assumed by DOE.
- The assumptions provided in Section 3 of this report with respect to the WMA C PA are validated.

The staff's primary review results related to Criterion A are as follows:

- DOE properly identified key radionuclides.
- DOE has a framework in place at Hanford to identify, evaluate, and implement different retrieval technologies to remove key radionuclide.
- DOE's approach to removing key radionuclides by implementing bulk waste removal to the limit of technology is appropriate in the absence of deployable technologies to target removal of key radionuclides in the tanks.
- DOE demonstrated removal of key radionuclides to the maximum extent technically and economically practical for the 100-series tanks, 200-series tanks, and most of the ancillary equipment.

- DOE did not demonstrate removal of key radionuclides to the maximum extent practical from plugged pipelines.

The recommendations provided in Section 2 (Recommendation #1 through #7) are collated into Table 5-1. None of these recommendations are categorized as (1) applicable to the Draft WIR Evaluation for WMA C. The recommendations are either (2) consider for future evaluations for waste management areas, or (3) general technical recommendations that would generally improve the basis for the technical information but are not essential to the evaluation for WMA C.

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

This section summarizes the information DOE submitted with respect to Criterion B of DOE Manual 435.1-1 (i.e., demonstrating compliance with the 10 CFR Part 61, Subpart C performance objectives) for the waste residuals remaining in WMA C tanks (and ancillary equipment), and documents the NRC staff's evaluation of that information. This section is divided into the different technical topics that the NRC staff evaluated.

The NRC staff review is divided into fifteen subsections. Those topics include the assessment context; future scenarios and conceptual models; climate and recharge; the engineered barrier system; the waste inventory; radionuclide flow and transport; biosphere characteristics and dose assessment; compliance with the performance objectives; model support; uncertainty, and quality assurance. These subsections are structured to summarize DOE's approach to the technical area in the Draft WIR Evaluation for WMA C (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 provided in each subsection are collated into Table 5-1, which provides a listing of all the recommendations in this TER, from Sections 2, 3, and 4. The recommendations are categorized as (1) applicable to the Draft WIR Evaluation for WMA C, (2) consider for future evaluations for waste management areas, or (3) general technical recommendations. Recommendations that are categorized as "consider for future evaluations" could be risk-significant depending on the specific details of the waste management area being evaluated, whereas, "general technical recommendations" are simply noted as best practices for performing waste evaluations. The recommendations are numbered in each subsection (e.g., Recommendation #8) and are indexed to Table 5-1.

Section 3.16 presents a summary of the NRC staff's overall conclusions for Criterion B. The NRC staff performed a risk-informed review, using the risk insights developed by DOE as well as independent analysis to complete the evaluation in this TER. The NRC staff used DOE's models as well as independent calculations. DOE developed its risk insights primarily from three sources of information: 1) a deterministic base case, 2) a probabilistic uncertainty analysis using a system model, and 3) safety function methodology using sensitivity analyses.

In the WMA C PA (DOE, 2016), DOE identified safety functions for WMA C as a feature of the system that provides a specific function that is relevant to the performance of the facility (e.g., I1 – Institutional control or EB3 – Steel shell permeability). DOE identified these safety functions in Appendix H of the WMA C PA. The NRC staff has included the safety functions DOE identified in the WMA C PA in the discussion in each relevant technical area. The NRC staff has summarized DOE's safety functions in Table 3-19 of this TER and has provided NRC's assessment of the risk significance of each safety function. The risk significance of the safety function or technical area is important to understand to provide context to the NRC staff's review of that technical topic.

The staff identifies many technical recommendations in the following subsections, however, as presented in Table 5-1, most of them are not applicable to the Draft WIR Evaluation for WMA C because of the low projected risks from WMA C waste residuals. These recommendations are relevant to future waste evaluations and determinations especially for sites with higher projected risks. The applicability of a recommendation is dependent on the risk-significance of the topic for the evaluation. Performance assessments can be complex models with numerous interdependencies and non-linear relationships between components or technical topics. Though the risk-significance of safety functions is provided in Table 3-19, the significance designation provided by the NRC staff is subjective and based on current understanding.

3.1 Assessment Context

This section evaluates the context of the WMA C PA (DOE, 2016), a term defined as "assessment context". The assessment context is essentially a description of the problem and systems involved in the PA that gives context around what is being analyzed. The assessment context includes the purpose of the PA, the regulatory framework, the overall assessment philosophy or strategy, provides the endpoints and timeframes for the assessment, and describes the waste characteristics and disposal system characteristics. In order to develop the assessment context of a PA, the following questions should be answered:

What is being assessed?
Why is it being assessed?
What is the scope of the assessment?

The purpose of conducting a PA may vary, as will the audience for the PA results. A well-defined assessment context can be used to determine the level of model abstraction as well as data and computational needs. The assessment context would also include the strategy to be used in the PA (e.g., conservative vs. realistic, simple vs. complex, deterministic vs. probabilistic). Because the assessment context sets the surrounding conditions for the PA, it is the first step in the PA methodology. Figure 3-1 shows the steps of the PA process.

3.1.1 Summary of DOE's Assessment Context

The regulatory framework component of the assessment context is within the HFFACO (The Tri-party Agreement). According to Section 2.5 of Appendix I of the Tri-party Agreement, the three parties (i.e., Washington State Department of Ecology, EPA, and DOE) elected to develop and maintain a PA as a tool to evaluate whether single-shell tank system closure conditions are protective of human health and the environment for all contaminants of concern, both radiological and non-radiological. As individual components are retrieved or characterized, or other component closure activities are completed, the resulting component characterization information will be incorporated into the WMA PA to determine its relative risk compared to the entire WMA performance. DOE identified safety functions associated with the assessment context. These safety functions were the extended institutional control timeframe (I1 - Institutional control), extended period of societal memory (I2 - Societal memory), probability of the point of calculation being the location of an actual future water well (I3 - Exposure), and the ability of the site's characteristics to minimize dose levels (S3 - Site characteristics). DOE also identified EB3 - Steel shell (permeability), EB4 - Steel shell (chemical), and EB15 - Pipelines (permeability) as safety functions although they are not part of the WMA C PA.

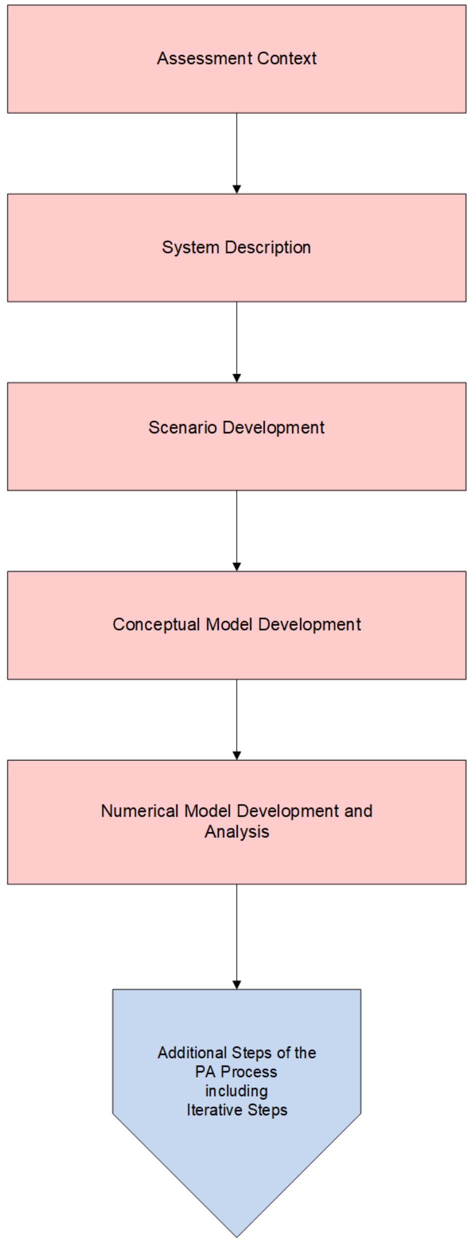


Figure 3-1 Initial Steps of the Performance Assessment Methodology

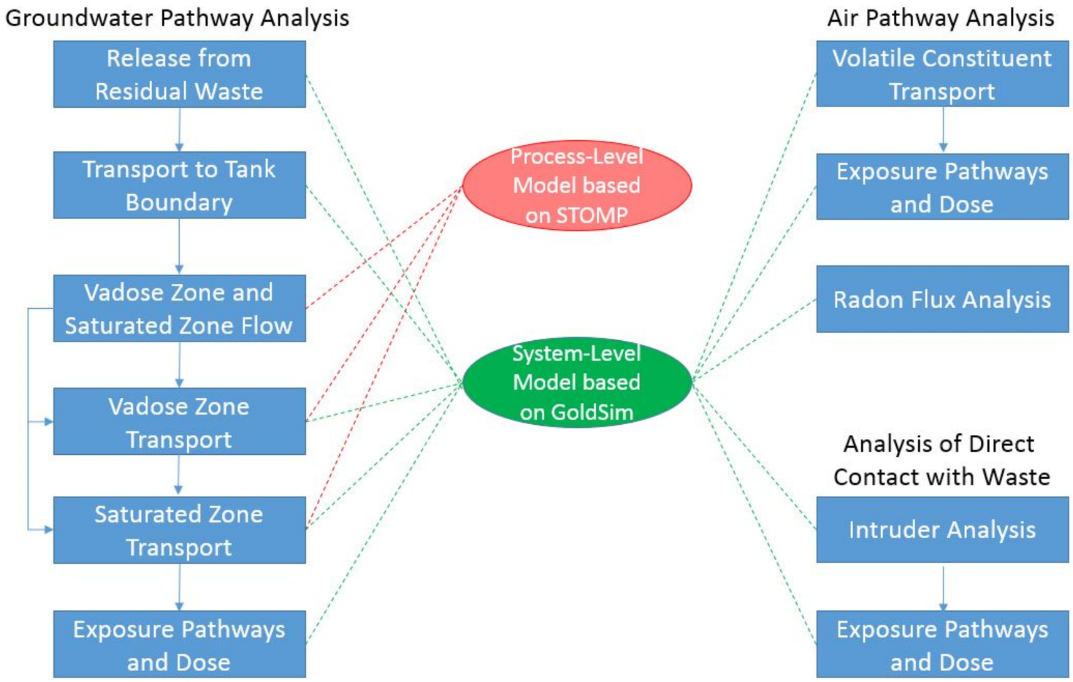
Final WMA closure decisions will be made after all components are retrieved and/or characterized, all other component closure activities have been completed, and a final WMA PA is completed. Based on the regulatory requirements outlined, the closure "performance assessment" as it is defined in HFFACO Appendix I will contain three major components. The PA for HFFACO is a broader analysis than a PA as defined in DOE Order 435.1. DOE, therefore, distinguishes between the term "Appendix I performance assessment" (IPA) when referring to the HFFACO Appendix I analysis and a PA as defined in DOE Manual 435.1-1.

The three major components of the IPA include: (1) a baseline risk assessment that evaluates human and ecological risks for current environmental contamination conditions, (2) an assessment based on the regulatory requirements of HFFACO Appendix I for hazardous constituents, and (3) a long-term PA on the fate and transport of radionuclide tank residuals in a tank farm. This third component is the WMA C PA. The PA is an evaluation of the potential impacts from radioactive residual waste remaining in tanks and ancillary equipment at the closed WMA C based on DOE Order 435. 1 (DOE, 2001b). The PA for WMA C should present a comprehensive, systematic analysis of the long-term impacts of a near-surface LLW facility after closure. The PA will also be used to support decisions related to radioactive waste resulting from the reprocessing of spent nuclear fuel, or WIR, that will be left at closure within tanks and ancillary equipment.

The overall assessment philosophy and strategy component of the assessment context includes the scope and analysis. The scope and analysis of the WMA C PA is designed to provide an informational basis for long-term human health and environmental information and assessments that will be needed by each regulatory body to approve eventual closure actions in a single-shell tank WMA. The metrics used for long-term performance results for the closed WMA C would be provided as an informational document and would not be compared directly to relevant regulatory standards. DOE will then compare and evaluate these performance metrics against appropriate regulatory standards in separate decision documents related to closure such as a Closure Plan. The PA will be maintained throughout the closure process under a PA maintenance program as required in DOE Order 435.1.

The modeling approach is also a component of the assessment context. DOE's modeling approach in the WMA C PA includes 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 are also evaluated. DOE provided a schematic representation of their overall modeling approach in the WMA C PA (DOE, 2016) as shown in Figure 3-2.

In the WMA C PA, DOE frequently used a hybrid approach to obtain simulated results which included both deterministic and probabilistic approaches. In the deterministic approach, the STOMP simulator process-based code (Subsurface Transport Over Multiple Phases (STOMP©) is copyrighted by Battelle Memorial Institute, 1996) was used in the analysis of post-closure flow and transport in the unsaturated and saturated flow systems and to examine a range of model parameters through sensitivity analyses. Additional transport analyses were carried out using a probabilistic approach, where the GoldSim-based system-level code (GoldSim© simulation software is copyrighted by GoldSim Technology Group LLC of Issaquah, Washington) was used to perform uncertainty analyses and additional sensitivity analyses to support the basis for comparisons with performance objectives under DOE Order 435.1. Modeling of source-term or contaminant release for grouted tank and ancillary equipment to the surrounding environment was also performed using a system-level model based on the software GoldSim, using its contaminant transport module. The source-term modeling considers mineral phase solubility-limited and matrix degradation rate-limited processes associated with release of contaminants



Subsurface Transport Over Multiple Phases (STOMP©) is copyrighted by Battelle Memorial Institute, 1996, and GoldSim© simulation software is copyrighted by GoldSim Technology Group LLC of Issaquah, Washington.

Figure 3-2 Overview of the Model Approach for the WMA C PA
[Figure 2-1 from the WMA C PA (2016)]

The source-term modeling considers mineral phase solubility-limited and matrix degradation rate-limited processes associated with release of contaminants from each of the 19 separate source terms. Radionuclide transport to the underlying unsaturated zone from the tanks and other structures filled with grout is via diffusion while for the pipelines transport is vie diffusion and advection. Once the contaminants reach the water table, deterministic and probabilistic modeling approaches are used to simulate contaminant fate and transport in the aquifer. Transport in the aquifer was simulated with the software STOMP© (DOE, 2016) to include significant features and processes influencing water flow and radionuclide transport in groundwater in deterministic simulations. In probabilistic simulations transport in the aquifer was simulated with GoldSim. Use of the FEPs analysis methodology for the WMA C PA was partially implemented to identify significant features and processes influencing flow and transport (RPP-RPT-41918, 2010). Probabilistic analyses for an abstracted model of the groundwater system, implemented in GoldSim, used probability density functions to represent the uncertainty in input parameters and demonstrate their influence on contaminant transport predictions. STOMP© flow fields for the unsaturated zone are used as inputs to the GoldSim-based model. Transport in the aquifer was simulated with the software STOMP (DOE, 2016) to include significant features and processes influencing water flow and radionuclide transport in groundwater in deterministic simulations. In probabilistic simulations transport in the aquifer was simulated with GoldSim.

Use of the FEPs analysis methodology for the WMA C PA was partially implemented to identify significant features and processes influencing flow and transport (RPP-RPT-41918, 2010). Probabilistic analyses for an abstracted model of the groundwater system, implemented in

GoldSim, used probability density functions to represent the uncertainty in input parameters and demonstrate their influence on contaminant transport predictions. STOMP flow fields for the unsaturated zone are used as inputs to the GoldSim-based model.

Modeling of the contaminant fate and transport through the air pathway was completed within the system-level model. Within the source-term model, inventory of radionuclides is partitioned between the aqueous and gaseous phases. Gaseous radionuclides can migrate by gaseous diffusion upwards to the ground surface and into the atmosphere. The WMA C PA models four gases in this way: ^{14}C as CO_2 gas, ^{3}H (tritium) as H_2 gas, ^{129}I as I_2 gas, and ^{222}Ra as radon gas. The GoldSim system-level model compares results with the performance objective of 20 pCi/m^2-s for radon flux at the surface of the disposal facility.

To meet the DOE Order 435.1 requirements, an all-pathways farmer exposure scenario is implemented to calculate the total effective dose equivalent for comparison to the performance objective of 0.25 mSv (25 mrem) and combines the dose from both the groundwater pathway and atmospheric pathway excluding the dose from radon and its progeny in air. The WMA C PA assumes the individual who receives dose from the groundwater pathway is a Representative Person (DOE, 2016) who resides near the C-Tank Farm and draws contaminated water from a well downgradient of WMA C. Water is used for drinking, irrigation of crops, and to water livestock. For the atmospheric transport pathway, air immersion, dust inhalation, and external exposure are dose pathways for the receptor residing 100 m (328 ft) downgradient of the facility fence line.

The characteristics of the waste and disposal system are additional components of an assessment context. Descriptive information relevant to the waste and disposal system at the WMA C Hanford Site are given in the WMA C PA, and provide the basis for conceptual models, (e.g., how radionuclides may be released following closure of WMA C). The WMA C PA includes topographic features and hydrogeologic characteristics which can strongly affect the fate and transport of contaminants potentially released from the closed site. Projected land use and population distributions can affect the estimation of impacts to humans. Facility features can control the release of contaminants and the rate at which they are released from the facility. The waste inventory, concentration of radionuclides, and volume and form of the waste can affect the magnitude and rate of radionuclide releases from the source term.

The assessment endpoints and timeframes are another component of the assessment context. The point of assessment and timing assumptions DOE uses follow requirements from DOE Order 435.1 and HFFACO. For example, institutional control and societal memory are assumed to last only for 100 years after the year of closure. Inadvertent human intrusion is assumed to occur after the active institutional control period. The intruder protection objective has been 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 100 m (328 ft) from the downgradient fenceline of WMA C per DOE Guide 435.1-1 (DOE, 1999a). Doses calculated for the all-pathways performance objective apply to a point of exposure 100 m (328 ft) downgradient of the facility fenceline (i.e., at the wellhead of a pumping well). Peak concentrations in groundwater are used as the concentration in the all-pathways analyses. Performance objectives and/or measures and the standards for all-pathways, atmospheric, radon flux, inadvertent intruder, and groundwater protection analyses are shown in Table 3-1.

The DOE Order 435.1 compliance time period for a PA is 1,000 years after closure; however, the WMA C PA also included a 10,000-year analysis period based on the recommendations in NRC's guidance document NUREG-1854 (2007) and to provide information to decisionmakers about potential long-term doses. DOE Manual 435.1-1 and DOE Guide 435.1-1 provide direction that a sensitivity-uncertainty analysis timeframe should include calculation of the maximum dose regardless of the time at which the maximum occurs, as a means of increasing confidence in the outcome of the modeling and increasing the understanding of the models used. Although DOE decided that the WMA C PA's 10,000-year analysis timeframe was sufficient to address uncertainty associated with radionuclides that impact groundwater during the compliance period, an additional evaluation case was run to evaluate the peak dose beyond the 10,000-year post-closure timeframe whereby a base-case version (with grout degrading at 30,000 years after closure) was run for a period of 400,000 years after closure.

3.1.2 NRC Evaluation of Assessment Context

The NRC staff reviewed various documents and sections of documents that described components of the assessment context and determined that the reviewed material adequately developed the context of the WMA C PA. The assessment context adequately describes the assessment purpose, regulatory framework, assessment philosophy, dose modeling methodology, compliance boundaries, assessment end points, and assessment timeframes. The NRC staff found information within the material that adequately addressed the questions that need to be answered in order to give context to the PA: what is being assessed, why is it being assessed, and what is the scope of the assessment? DOE defined the level of model abstraction, as well as data and computational needs and the strategy for the PA.

The NRC staff has determined that the following safety functions are relevant to the safety of the facility: I1 – Institutional control; I2 – Societal memory; I3 – Exposure; and S1 – Site characteristics. However, these safety functions are not the same type of safety functions as others listed in Table H-1 in the WMA C PA (2016) and are outside the assessment context. Section H.2 describes how the safety functions are features of the system that provides a specific function. The three "institutional" safety functions (i.e., I1, I2, and I3) are events and not features or barriers that can be evaluated for performance. As previously discussed, these safety functions allow either the institutional control period to end at 100 years after closure or not; extend the period of societal memory past 100 years or not; and allow the hypothetical well and therefore the point of calculation to be located 100 m (328 ft) downgradient of the site's fence line or not. In order to have consistent PA results, rules or regulations were established that dictate how DOE will calculate the output using the PA. In section H.3 DOE stated that the "The goal of the PA is to evaluate these safety functions, to provide reasonable assurance of performance even when some of the safety functions are lost or degraded through time or disruptive events." This cannot be done with the three "institutional" safety functions. Safety function S1 (site characteristics) is a set of many features related to the characteristics of the chosen site. These features are listed later in Table H-1 of the WMA C PA and mostly accounted for under the unsaturated zone or the saturated zone safety functions. If there are features or processes unaccounted for (e.g., precipitation rate), they should be listed separately. Also, safety function SZ4 [saturation zone 4] - Dilution in well, is predetermined by regulation and not evaluated in the PA. In addition, safety functions EB3 - Steel shell (permeability), EB4 - Steel shell (chemical), and EB15 - Pipelines (permeability) were not part of the PA by choice of the DOE, and therefore not relevant to the outcome of the PA although it is relevant to the

Table 3-1 Performance Objectives and Standards used in the WMA C PA
[Figure ES-1 in the WMA C PA (2016)]

Performance Objective and/or Measure	Standard
All Pathways (DOE Order 435.1 Chg 1)	0.25 mSv/yr (25 mrem/yr) EDE
Atmospheric (40 CFR 61, Subpart H)	0.10 mSv/yr (10 mrem/yr) EDE
Atmospheric (40 CFR 61, Subpart Q)	20 pCi/m^{-2}-s (1.9 pCi/ (ft^2-sec)) radon flux (at surface of disposal facility)
Acute Inadvertent Intruder (DOE Order 435.1 Chg 1)	5.0 mSv (500 mrem) EDE
Chronic Inadvertent Intruder (DOE Order 435.1 Chg 1)	1.0 mSv (100 mrem/yr) EDE
Groundwater Protection (water resources) (40 CFR 141)	Beta-gamma dose equivalent ≤ 0.04 mSv/yr (4 mrem/yr)
	Gross alpha activity concentration (excluding radon and uranium) ≤ 555 Bq/m^3 (15 pCi/L)
	Combined Ra-226 and Ra-228 concentration ≤ 185 Bq/m^3 (≤ 5 pCi/L)
	Uranium concentration ≤ 30 µg/L (≤ 0.03 ppm)
	Sr-90 concentration ≤ 300 Bq/m^3 (8 pCi/L)
	H-3 concentration ≤ $7.4 \times 10^{+5}$ Bq/m^3 (≤ 20,000 pCi/L)

safety of the facility. Information or technical bases were provided that support their capability as safety functions.

The NRC staff recommends 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), as a means of increasing confidence in the outcome of the modeling and allowing stakeholders and others to know approximately where doses will peak in case doses are rising at the end of a 1,000-, 10,000-, or 400,000-year timeframe. Clearly, doses after 1,000 years need not be directly compared with performance objectives and measures provided in the DOE Order (Recommendation #8).

Summary of Review

- The NRC staff reviewed various documents and sections of documents that described components of the assessment context and determined that the reviewed material adequately developed the context of the WMA C PA.

- There no risk significant FEPs or major sources of uncertainty associated with this section.

- Recommendation #8 is discussed in this section and is included in Table 5-1 of this report.

3.2 Future Scenarios and Conceptual Models

Uncertainties must be evaluated within a PA process and can involve separate treatments of scenario uncertainty (future uncertainty), model uncertainty, and parameter uncertainty (NRC, 2015). Uncertainty about the future of the site is the result of inherent lack of knowledge about how the site will evolve over time. Scenario development is a commonly used technique to account for the potentially large uncertainty associated with the future since the future may include potentially disruptive events such as an igneous or climate event. Model uncertainty encompasses the uncertainty in the conceptualization of the system, the uncertainty in its mathematical representation, and the uncertainty in the solution of the mathematical representation (Bonano and Cranwell, 1988). Conceptual model uncertainty is frequently the dominant type of uncertainty in a PA due to limitations in the available supporting data. The conceptual model should be based on the information and data available and consider significant features, events, or processes (FEPs) to include all plausible representations of different ways a disposal system might behave (i.e., alternative conceptual models). Uncertainty with regards to the range of data values, or parameter uncertainty, can be propagated through the PA by distributions of variables.

Formal approaches to scenario development are usually either known as “bottom-up” or “top-down.” Both approaches should be able to capture the features and phenomena that are potentially relevant to near- and long-term performance of a disposal system. For the bottom-up approach, the FEPs analysis developed should produce a list of features that are present at the disposal site and facility, processes that occur or will occur at the disposal site and facility, and plausible events. Events are usually abrupt changes to the disposal site or facility that have the potential to affect the performance of the disposal system (e.g., earthquakes, floods, storms, volcanic eruptions). From this set of potentially relevant FEPs, a subset of FEPs can be defined that are used to identify a probable future evolution of the disposal site (i.e., a central scenario). Usually, the central scenario does not include disruptive events while alternative scenarios frequently include disruptive events (NRC, 2015). However, in rare cases, if disruptive events are expected to occur during the assessment period, then they are typically included as part of the central scenario. Plausible conceptual models describe the behavior of the system (e.g., vertical downward flow in the unsaturated zone and lateral flow in the aquifer). Capturing all or most of the plausible behaviors on how the system may function within the central scenario will reduce conceptual model uncertainty. A qualitative description of the conceptual model would include how the FEPs and significant barriers interact with one another and how the site functions. Although the distinction between a scenario and a conceptual model may occasionally overlap during the PA process, it is important that an attempt be made to capture the full range of possible future states of the disposal system and associated conceptual models.

For the top-down approach, safety functions are used to develop scenarios. A safety function is defined qualitatively as a function through which a component of the disposal system contributes to safety and achieves its safety objective throughout the timeframe of the assessment. That is, a safety function is a feature of the system that provides a specific function that is relevant to the performance of the facility.

The purpose for including plausible alternative scenarios and plausible alternative conceptual models within a PA is not to replace the central scenario or a base case conceptual model, but to reduce overall uncertainty and to inform stakeholders and those making decisions on potential risks based on probability and consequence. Implausible or what-if models or scenarios need not be part of the decision-making process since they should have been excluded during the scenario and conceptual model development although sensitivity analyses can include such implausible parameter ranges in order to gain insights into the behavior of the disposal system. However, emphasizing "what-if" conceptual models or scenarios has the potential undesirable effect to shift the focus the attention of decisionmakers or stakeholders to cases that have low consequence or very low probability or both (i.e., cases with very low risk). See NRC (2015) for more detailed information on scenario and conceptual model development.

3.2.1 Summary of DOE Analyses of Future Scenarios and Conceptual Models

DOE evaluated alternative conceptual models as part of their sensitivity analyses. Alternative geologic models were used to examine the potential impacts of lateral flow in the unsaturated zone, a concern for many stakeholders.

3.2.1.1 The Central Scenario and Alternative Scenarios

Most FEPs with a potential to alter the long-term evolution of the site, and with sufficient probability of occurring, were analyzed within the uncertainty and sensitivity analyses. Based on the uncertainty/importance analysis and sensitivity analysis of those results, and the analysis involved in producing the list of FEPs in Appendix H in the WMA C PA, DOE concluded that there were no plausible alternative future scenarios for the region around WMA C and that no other scenario would need to be evaluated besides the central scenario. That is, no disruptive events or processes were identified that could significantly change the long-term evolution of the site (e.g., igneous intrusion or volcanic ashfall, earthquakes, or major changes to the hydrologic or hydrogeologic system). The single scenario that the WMA C PA evaluates assumes a relatively unchanging natural system at WMA C and relatively slow changing, or degrading, engineered system (i.e., engineered surface barrier, structures and components out of cementitious material) at WMA C.

DOE used a hybrid approach as described in the WMA C PA (2016) for identifying future scenarios at the Hanford Site and relied on the top-down approach. The ability of each FEP to affect the safety functions is identified by DOE and documented in Table H-1 of the WMA C PA. DOE performed sensitivity analyses on specific safety functions. Each safety function may be associated with several FEPs that influence the system in a similar manner and possibly represent an aggregated view of the potential negative effects of the FEPs. For example, FEPs that have the potential to degrade cementitious material, such as the grout or the basemat, were tested together in a sensitivity analysis. That is, the specifics of what causes the FEP to accelerate degradation of cementitious material was not relevant (e.g., physical or chemical processes). Multiple FEPs were evaluated with relatively few sensitivity cases, and, according to the DOE, key issues and significant safety functions were the focus of the assessment. In addition, DOE concluded that this approach avoided a large amount of effort without a commensurate improvement in the PA. If a safety function being analyzed showed minimal sensitivity to performance or dose, FEPs associated with that safety function were not likely to be sufficiently significant to be part of an alternative future scenario for the site or an alternative conceptual model.

Chapter 8 of the WMA C PA presents results of the uncertainty/importance analysis and sensitivity analysis. The results of the uncertainty/importance analysis are discussed in Section 8.1.5.1 of the WMA C PA. Each individual calculation of a probabilistic model is called a realization. One DOE analysis ran 300 realizations and correlated observed changes in the peak dose with changes in the sampled values of the input parameters. The peak dose values for the 300 realizations varied within about two orders of magnitude range ($1x10^{-3}$ to $1x10^{-2}$ mSv/yr [0.01 to 1 mrem/yr]). These results indicate that parameters associated with the saturated zone, unsaturated zone, and ^{99}Tc inventory were the most important to the output. However, Section 8.1.5.1 also stated that hydraulic properties associated with the unsaturated zone flow may have lower influence on affecting peak dose than the results might indicate. The results of the sensitivity analysis were presented in various tables in Section 8.2 of the WMA C PA. Table 8-15 provides a brief explanation of the range of parameter values used within each safety function. The safety functions tested within the sensitivity analysis included recharge through the engineered surface cover, inventory estimates, release function of the residual chemistry, diffusional coefficient, and advective release, and properties associated with the saturated zone, the unsaturated zone, and cementitious material. Using maximum concentration at the downgradient point of calculation (POC) as a measurement of sensitivity, no sensitivity cases resulted in any increases greater than 4.5 of the maximum concentration of the base case. That sensitivity case involved using the upper bound values of the inventory estimate, which increased the maximum concentration of the base case from $1.2x10^3$ Bq/m^3 to $5.3x10^3$ Bq/m^3 (32 pCi/L to 144 pCi/L). Most of the safety functions were tested in sensitivity cases by changing the range of values for one or two parameters associated within the safety function.

Most FEPs with a potential to alter the long-term evolution of the site, and with sufficient probability of occurring, were analyzed within the uncertainty and sensitivity analyses. Based on the uncertainty/importance analysis and sensitivity analysis of those results discussed above, and the analysis involved in producing the list of FEPs in Appendix H in the WMA C PA, DOE concluded that there were no plausible alternative future scenarios for the region around WMA C and that no other scenario would need to be evaluated besides the central scenario. That is, no disruptive events or processes were identified that could significantly change the long-term evolution of the site (e.g., igneous intrusion or volcanic ashfall, earthquakes, or major changes to the hydrologic or hydrogeologic system). DOE's central scenario for the long-term evolution of the site can generally be described as a continuation of the features and processes as they currently exist or as currently designed to exist (e.g., tank basemats, infill grout) for thousands of years and involve no disruptive events. To exclude seismic activity as an event with the potential to create an alternative scenario, a sensitivity case was run where the grout properties were changed from being relatively impermeable to a much more permeable end state. To emulate earthquake-induced fractures the tank structure and the embedded grout are given the hydraulic properties of sand at differing times after closure. The results of this sensitivity analysis indicated relatively minimal changes (i.e., less than a factor of 2) in the maximum concentration. Changes in climate that involved increased long-term rainfall were assessed by increasing the rate of recharge and these also produced relatively minimal changes (i.e., less than a factor of 2) in the maximum concentration.

3.2.1.2 The Conceptual Model and Alternative Conceptual Models

The main conceptual model is an analysis case that has been labeled by DOE as the base case. It involves safety functions performing as expected with no unanticipated disruption, although a probabilistic barrier importance analysis of the base case was conducted. This was to demonstrate the effects of parameter uncertainty on system performance where parameters were assigned probability density and uncertainty estimates in dose were evaluated. In addition, a set of deterministic sensitivity analyses demonstrated the effects on radionuclide concentrations or dose from safety functions that were degraded, in contrast to the expected safety function behavior as defined in the base case. The base case assumed a closure of the WMA C in 2020 with a 100-year institutional control period. The DOE Order 435.1-defined compliance time period is 1,000 years after closure. The yet-to-be designed engineered surface cover is assumed to be fully functional for the first 500 years and fully degraded after that. Infill grout and concrete tank components were assumed to remain intact and fully functional for 30,000 years.

Residual inventory estimates in the tanks and ancillary equipment were estimated by various means and based on information and conditions as of September 2014 with a total of 43 radionuclides being evaluated in the WMA C PA. Highest inventory estimates were for the 16 single-shelled tanks, the CR-vault tank, the catch tank, and the pipelines, which are all planned to be filled with grout except for the pipelines. Residual waste from pits and diversion boxes are incorporated as part of the pipeline source term. Radiological contaminant releases from the intact grout are controlled by diffusion processes while contaminant release from wastes within the pipelines assumes a release by diffusion and advection. The main contaminant release mechanisms from the waste include mineral phase solubility-limited (e.g., uranium isotopes) and matrix degradation rate-limited processes (e.g., ^{99}Tc). The major contaminant pathways whereby radionuclides are transported are by groundwater, air, and human disturbance by inadvertent drilling through the waste. The main conceptual model for the groundwater pathway begins with precipitation, followed by a small rate of water infiltrating through an engineered surface cover and a greater rate of infiltration when the cover is degraded, infiltrating water then contacting waste, and released radionuclides transported through the infill grout and intact concrete tank wall and basemats. Contaminants entering the unsaturated zone move at a rate downward dependent on the rate of infiltrating water and eventually reach the saturated zone that has an assumed thickness, hydraulic gradient, hydraulic conductivity, and direction of flow during the 1,000-year compliance period. Once the contaminants reach the water table of the aquifer, they mix with that water and become diluted as they travel down 100m (328 ft) from the fence line of the WMA C. The WMA C PA estimates the groundwater dose to a hypothetical member of the public living at this location who consumes contaminated groundwater, leafy vegetables, produce that were irrigated with contaminated groundwater, and contaminated milk and meat.

DOE assessed two alternative conceptual models of the WMA C system: Alternative Geologic Model II and a heterogeneous media model. Alternative Geologic Model II represented an alternative conceptualization of the geologic model, and not a parametric variation. The stratigraphic and geologic differences between the base case conceptual model and the Alternative Geologic Model II is shown in Table D-7 in the WMA C PA (2016). The Hanford H2 sand is divided into three separate units, where the upper two-thirds is sand, but the lower third is given properties of a coarse sand followed by a silty sand layer at the bottom of the H2 subunit. The basis for the alternative conceptual model is gross gamma and potassium data indicating a coarsening of the sand in the lower part of the H2 so that it is represented by a sandy gravel (DOE, 2016). Wet sieve particle size distributions for borehole 299-E27-22 sediments presented in Table B-4 of the WMA C PA also indicated this interpretation. Data supporting a silty sand layer at the H2 bottom were strong potassium peaks and an occasional but strong natural uranium peak. In discussions with stakeholders and regulators, and with the geologic interpretation prepared with input from the technical staff of the Nez Perce Tribe, the Alternative Geologic Model II was developed and incorporated into the PA as a means to explore the performance implications of the alternative conceptualizations, specifically a potential for increased lateral movement in the unsaturated zone due to the different soil hydraulic property values.

The heterogeneous media conceptual model was developed as an alternative to the equivalent homogeneous medium (EHM) model which was used as the base case in the PA where each heterogeneous unit is assigned an upscaled hydraulic properties (i.e., small, core-scale measurements were upscaled to larger field-scale properties). For the heterogeneous media model, the natural moisture content distribution is an indicator of sediment texture and thus, soil hydraulic properties. With higher moisture contents associated with fine-textured sediments and lower moisture contents associated with coarse-textured sediments, a geostatistical interpretation of the moisture data was used to identify and select hydraulic properties for input into an unsaturated zone model for WMA C and to develop an alternative unsaturated zone conceptual model. The results are compared to identify the impact of heterogeneity on predicting concentrations in the water table.

A third conceptual model of the geology was analyzed although it was classified as being implausible by DOE. DOE stated that a large clastic dike is unlikely to exist within WMA C because no neutron or drywell moisture measurements have detected evidence of a continuous band of high moisture (Section 8.2.3 in the WMA C PA (2016)). The clastic dike conceptual model included the representation of a preferential pathway, such as a clastic dike or unsealed borehole, located underneath Tank C-105. The sensitivity analysis performed by DOE involved creating and evaluating this alternative conceptual model by modifying Alternative Geologic Model I and incorporating two large clastic dikes under several of the WMA C tanks. The clastic dikes extended down through most of the unsaturated zone and the length and width of WMA C. The hydraulic parameters assigned to the clastic dike material were selected to provide high pore-water velocities and to determine whether the flux conditions exist at WMA C, such that the clastic dikes provide a preferential flow path for the residual waste.

3.2.2 NRC Evaluation of Future Scenarios and Conceptual Models

While the staff has some recommendations related to DOE's approach, NRC staff finds that DOE has adequately developed appropriate conceptual models and scenarios for the waste residuals at WMA C due to the overall safety margins in the PA results analyzed, including uncertainty and sensitivity analyses. For more risk significant tank farms and waste management areas at the Hanford Site, these issues would require additional evaluation.

NRC staff has concluded that the central future scenario being used in the WMA C PA is plausible; however, notes that a single future scenario for a 1,000-year compliance period or 10,000-year timeframe requires a robust technical basis. Similarly, NRC staff has concluded that most components of the main conceptual model are plausible and have robust technical bases; however, DOE's documentation of the results of the safety function methodology do not describe how alternative conceptual models were identified for evaluation (i.e., determined their plausibility). NRC staff considered other alternate future and conceptual models which were not specifically identified and eliminated from consideration in the original documentation by DOE. For example, alternative future scenarios pursued by NRC staff centered on the future river water levels of the Columbia River influenced by the future existence of dams along the river or climate conditions upstream in the Columbia River watershed. However, even if groundwater flux dilution was decreased by a factor of 110 (11,000 m/d ÷ 100 m/d), peak concentration or dose objectives would be met based on given base case results. NRC staff also considered alternative conceptual models that were centered on water bypassing the infill grout due to concrete degradation or other processes. A comment response provided by DOE (DOE, 2019) included a combined sensitivity case presenting dose results from increasing the hydraulic property of all the tank components, including the pre-closure concrete with rebar, so that advective flow is present in and around the tanks. Results indicated that the performance objective would be met.

DOE does identify and document an extensive list of relevant FEPs (see Appendix H in the WMA C PA) and identified those FEPs that may degrade or modify the performance of a safety function in some way. However, in NRC (2019), NRC staff asked how this hybrid approach identified safety functions that influenced one another or identified the interdependencies and interrelationships between the identified features and phenomena. DOE stated that, "multiple FEPs acting on a single safety function represent an interdependency, in which the multiple FEPs may result in qualitatively similar type of degradation but may increase the rate at which degradation occurs" (DOE, 2019). The NRC staff disagrees with this statement since there are numerous instances where multiple FEPs acting on a single safety function are not dependent on one another.

Using DOE's example in DOE (2019), two FEPs potentially deleterious to the safety function (EB9 – grout degradation) were given as FEP 1.1.08 "Quality Control" (defects during construction causes grout to crack) and FEP 1.2.03 "Seismicity" (earthquakes cause cracks/openings in the grout). The two FEPs may cause similar degradation to the grout and both FEPs may cause the rate of degradation to increase compared to the occurrence of only one FEP, but there is no interdependency in terms of occurrence between the two, although a poor-quality concrete would likely be more susceptible to cracking from an earthquake. One FEP may be present without the other; neither is dependent on the other. In addition, both example FEPs are associated with more than just cracks/openings in the grout. FEP 1.1.08 "Quality Control" can cause general defects in construction of a disposal system, improper or faulty waste emplacement and backfilling, defects during the conditioning of the waste, or defects in the cover construction. FEP 1.2.03 "Seismicity" can cause changes in the physical properties of rocks due to stress, hydrological changes, faulting, or soil liquefaction. Since more than one of these changes could cause changes to the rate of contaminant release, one sensitivity analysis cannot exclude either FEP from further evaluation. For example, an alternative conceptual model involving the quality control FEP may include a crack-free grout with alternative hydraulic and/or chemical properties due to poor quality grout that affect model output. If, on the other hand, during the FEP analysis, a sufficiently sound technical basis demonstrated low probability, no additional evaluation of that FEP would be necessary (NRC, 2015).

DOE also stated (DOE, 2019) that, a:

> “FEP that is relevant to more than one safety function indicates a second type of interdependency identified in the approach. A potentially deleterious FEP that applies to more than one safety function indicates the potential for a common failure mechanism. For instance, in the example discussed above, seismicity has the potential to affect both the grout hydraulic safety function, and the tank structure safety function. Therefore, a sensitivity case developed to address this situation should take account of this potential for common failure.”

The NRC staff agrees that a potentially deleterious FEP that applies to more than one safety function indicates the potential for a common failure mechanism (e.g., an earthquake event fractures both infill grout and tank structure). The NRC staff disagrees that this is a second type of interdependency. A FEP may affect one or more safety functions due to a common failure mechanism, but the FEP remains a single FEP and there is no interdependency. Using FEP 1.1.08 “Quality Control” as an example for the interdependencies between FEPs and/or safety functions, a failing quality control could result in an unforeseen poor-quality grout that has different hydraulic and/or chemical properties than the original technical specifications. This difference, in turn, may affect safety function EB10, or the chemical properties of the grout that can beneficially affect the chemistry of the waste residuals. An increased water flow and a reduced pH environment could then, in turn, affect safety function WB1 “Residual Waste (chemical)” resulting in quicker contaminant release. The sensitivity and uncertainty analyses in the WMA C PA do not allow DOE to identify interdependencies and interrelationships between FEPs that could result in plausible alternative conceptual models or alternative future scenarios.

As previously discussed, overall uncertainty can be divided into three parts: scenario, model, and parameter uncertainty. One-at-a-time sensitivity cases, as used in the WMA C PA, do not lend themselves to identifying risk-significant interdependencies and interrelationships between FEPs that can lead to plausible alternative scenarios or conceptual models. The WMA C PA uncertainty analysis was focused on the evaluation of the range of variability of the input parameter values for the base case in addition to identifying important barriers. The uncertainty analysis was not suited to identify plausible alternative conceptual models. Sensitivity case “GRT4” is discussed in Section 3.4.1.2.1 but will also be briefly discussed in this section, since it is a good example that individual sensitivity runs are unable to capture the interrelationships between FEPs. Despite the degraded cementitious material, the original GRT4 case showed a relatively modest 144% change to the maximum concentration at the downgradient POC, potentially indicating that the grout and tank did not provide the isolating capacity that was originally assumed and that the flow and transport properties of the infill grout and concrete shell were not risk significant. The NRC staff’s RAI 2-9 (NRC, 2019) pointed out that because of the assumed property values of the Hanford H2 sand, the permeability contrast between the cementitious material and the surrounding backfill material created a barrier to hydraulic flow. DOE responded (DOE, 2019) by modifying the sensitivity case GRT4 and changing the assumed hydraulic property values of the grout and concrete of the tank to be similar to those of the surrounding gravel-dominated backfill material. This modified sensitivity case increased the maximum concentration at the downgradient POC for the base case value from 30 pCi/L to 640 pCi/L ($1.1x10^3$ Bq/m^3 to $2.4x10^4$ Bq/m^3), or an increase by a factor of 22.

Returning to the example FEP 1.2.03 Seismicity in DOE (2019), DOE had stated that the overall effect from seismic activity on the system would be increased permeability of the cementitious material, and therefore, could be addressed by a sensitivity analysis evaluating the effect of increased permeability of the grout. However, DOE did not appear to be aware of the interrelationship between hydraulic conductivity, waste release, and additional FEPs or processes, like capillary action, so that a review of the sensitivity case "GRT4" results in isolation indicated that changes to the hydraulic conductivity were not significant. A completed FEPs analysis would have identified such interdependencies and interrelationships between FEPs and avoided the masking of the direct sensitivity of the permeability of grout and concrete on radionuclide concentrations.

The NRC staff recommends that for future WIR evaluations and assessments, DOE's hybrid approach to scenario and conceptual model development, with its emphasis on the top-down method and the use of uncertainty and sensitivity analyses, should be refined since DOE's safety function methodology does not appear to be able to identify significant interdependencies and interrelationships between FEPs that could result in plausible alternative future scenarios or alternative conceptual models (Recommendation #9).

The NRC staff recommends that for future WIR evaluations and assessments, the analysis and results of the safety case approach, including potential alternative conceptual models that were excluded or included in the assessment, or the technical basis for assessing a single future scenario, be clearly described and documented (Recommendation #10).

3.2.2.1 NRC Evaluation of the Central Scenario and Alternative Scenarios

While this section will evaluate if the uncertainty associated with future scenarios has been bound by the WMA C PA, the specific disruptive processes and events that may cause significant long-term changes at the site are reviewed in the respective topical sections of this report. For example, climate is reviewed in Section 3.3, while geologic stability is reviewed in Section 3.12. DOE's central scenario for the long-term evolution of the site continues the features and processes as they currently exist (e.g., climate, vegetation, vadose zone, aquifer), or as currently designed to exist (e.g., tank basemats, infill grout), for thousands of years and involve no disruptive events. No analysis or statement was provided based on the results of the safety case demonstrating that alternative scenarios were not needed to provide confidence in the PA results. Although, NRC staff agrees that DOE's central scenario is very plausible, DOE's safety function methodology does not appear to be able to identify interdependencies and interrelationships between FEPs that could result in plausible alternative future scenarios. NRC staff considered other alternate futures which were not specifically identified and eliminated from consideration in the original documentation by DOE.

An example of a feature that may lead to an alternative future scenario is the removal of Columbia River dams such as the Priest Rapids Dam and other large dams upstream of the Hanford Site. Although impacts to groundwater systems do occur, research on dam removal typically focuses on modifications to the river system. Dam emplacement or removal actions modify adjacent groundwater system boundary conditions and often result in a rise or fall in the underlying and adjacent water table (Berthelote, 2013). Berthelote (2013) documented the results of an analysis that predicted the changing position of an underlying water table after removal of an 8.5 m (28 ft) high dam in western Montana. The simulated water table declines compared favorably with the observed declines. In addition, river stage fluctuations can create changes in nearby hydrogeological systems. The WMA C PA (2016) states on page 8-30 that, "Even though the hydraulic gradients are likely to remain very small (around 10×10^{-5} m/m) as the water table declines in the future, current monitoring has indicated that gradients can vary by a factor of two, due to Columbia River stage fluctuations and interconnections to the aquifer in the Central Plateau." Possible effects of removing dams from the Columbia River upstream from the Hanford Site may include changes to the direction of flow, hydraulic gradient, water table, or river stage fluctuations.

NRC staff asked DOE in (NRC, 2018a) what the impact of the upstream dams has on subsurface hydrology at the site and what the long-term plans are for the dams. DOE stated that the assumption within the PA was that the dams would continue to be there for the length of the PA modeling period and that, if a dam is removed, it would be replaced. In addition, DOE believes that, due to the current state of information, the best scenario for the base case is a long-term, steady-state hydraulic gradient for the 200 East area. However, DOE did not provide a technical basis for the assumption of the continued existence of Columbia River dams for the length of the compliance period. The possibility exists that maintenance of ageing dams and continual sedimentation removal will become cost prohibitive. An assumption that there is only one plausible future scenario (i.e., no alternative future scenarios) with a non-dynamic environment for 10,000 years requires a rigorous technical basis. Since the existence or absence of dams upstream of the Hanford Site is a potential disruptive event, there is a sufficient basis to assess an alternative scenario with regards the absence of dams upstream of the Hanford Site.

DOE indicated in NRC (2018a) that in the PA, DOE looked at a range of groundwater fluxes, which is the primary variable of interest with respect to dilution of radionuclides entering the saturated zone from the overlying unsaturated zone. NRC staff agrees that if major changes to the site, or the hydrogeologic regime, were limited to an increase or decrease in the hydraulic gradient, sensitivity analyses DOE performed were sufficient to gain the risk insights needed. However, as previously mentioned in Berthelote (2013), water table levels could also be affected such that the groundwater flow or discharge (i.e., flux times area) at the WMA C may increase or decrease. In addition, DOE had indicated that a post-closure PA can still be protective of safety without accurately representing potential future behavior of the system, if unambiguously conservative assumptions are made that clearly bound the potential effect of any deleterious FEPs on the safety function. However, for the groundwater flux ranges used in sensitivity and uncertainty analyses, NRC staff discusses in Section 3.7 that ranges DOE selected are not unambiguously conservative. The sensitivity of groundwater volumetric flow or discharge was not part of the analyses as documented in Sections 8.1.3.6 and 8.2.2 in the WMA C PA (2016). An additional alternative scenario, or subset of the potential alternative scenario discussed above, that may be plausible would involve long-term drought in the upper reaches of the Columbia River watershed lasting two decades or longer (e.g., a major snowfall reduction in the Canadian Rockies). In combination with an unchanged climate pattern in the Columbia Plateau, such a scenario would see lower water levels in the Columbia River and the groundwater while average long-term precipitation rates at the Hanford Site remain consistent.

3.2.2.2 NRC Evaluation of the Conceptual Model and Alternative Conceptual Models

This section will evaluate if the uncertainty associated with conceptual models has been bound by the WMA C PA. NRC staff has concluded that most components of the main conceptual model, or the base case, are plausible and have robust technical bases; however, since DOE's documentation of safety function methodology does not describe how alternative conceptual models were identified for evaluation (i.e., determined their plausibility) or able to identify all interdependencies and interrelationships between FEPs that could result in plausible alternative conceptual models, NRC staff considered a number of alternative conceptualizations which were not specifically identified and eliminated from consideration in the original documentation by DOE. As described in Section 3.2.1.2, the base case involves many components of the disposal system including components from the engineered system and the natural system. These components of the main conceptual model are evaluated in the respective section depending on the topic (i.e., components of the main conceptual model that involve the basemat are evaluated in the Section 3.4 and the base case conceptualization of flow and transport through the saturated zone are discussed in Section 3.7 of this report). For example, the three identified alternative conceptual models of flow and transport through the unsaturated zone were evaluated in the WMA C PA. While these alternative conceptual models are discussed here, they are evaluated and discussed in detail in Section 3.6 on unsaturated zone.

DOE's WMA C PA has evaluated alternative conceptual models, such as the Alternative Geological Model II and the heterogeneous media model, which provided useful information on complicated issues related to the unsaturated zone. DOE has stated, and NRC staff agrees, that that these two alternative conceptual models are plausible. The clastic dike alternative model was also evaluated in the PA, although it was considered implausible and included as a "what-if" and there is no evidence of dikes at WMA C (NRC, 2018e). In general, the results of the safety function methodology were not clearly documented, and plausible alternative conceptual models that were assessed were developed independent of the safety function methodology.

The heterogeneous media model was developed to account for subsurface heterogeneities in greater detail. Previous studies had indicated that moisture content may be an indicator of sediment type and different sediments impact subsurface transport (RPP-CALC-60345, 2016). The alternative conceptual model was to provide confidence that geologic heterogeneities do not impact contaminant transport behavior significantly enough to alter the risk assessment for closure of WMA C (Freedman et al., 2019). NRC staff encourages DOE to emphasize the conceptual model development aspect of the PA, especially for future analyses of more risk-significant sources at the Hanford site, since an uncertainty/barrier importance analysis and a single parameter sensitivity analysis is not a substitute for the plausible ways a disposal system may function. Developing plausible conceptual models to refute or support FEPs will reduce conceptual model uncertainty for future assessments, which is the dominant type of uncertainty in a PA due to limitations in the available supporting data.

NRC staff identified and discussed in this report areas where relevant analyses or information could reduce uncertainty and provide additional confidence in the PA results. NRC staff concluded that the risk-significance of separate safety functions associated with pre-closure cementitious material and rebar is not known, since all cementitious features, such as infill grout, concrete basemat, walls, and the dome were assumed to degrade together at the same rate and sensitivity analyses were not performed that tested the safety functions separately. Because DOE analyzed the safety functions together in one group, alternative conceptual models associated with near-field flow arising from differential rates of degradation could not be identified. NRC staff encourages DOE to test plausible alternative conceptual models for future WIR evaluations and assessments based on the spatial and temporal performance of each safety functions related to features with cementitious material. These include EB8 - Tank structure (permeability); EB9 - Grout in tank (permeability); and EB13 - Tank Base Mat (permeability).

A plausible alternative conceptual model may arise from the uncertainty associated with pre-closure concrete such as: the consistency of the concrete quality produced during the construction of the C-Tank Farm, the placement of concrete during that construction, potential rebar corrosion byproducts accelerating concrete degradation, and the lack of data with regards to the current conditions of basemats. Currently, the risk significance of the engineered surface cover performance as an infiltration barrier is ranked low. The engineered surface barrier may increase in importance in an alternative conceptual model whereby infiltrating rainwater flows in the degraded concrete dome, wall, and basemat along the side of an intact grout.

An additional alternative conceptual model, or subset of the potential alternative conceptual model discussed above, that may be plausible would involve preferential flow along shrinkage gaps between the infill grout and the steel liner or concrete wall (discussed in Section 3.4.1.2.2). In Sections 5.1.6 and 5.6.2 in RPP-RPT-46879 (2016), DOE briefly discusses shrinkage of concrete as it cures and after it cures. Dinwiddie et al. (2013) investigated the potential of curing grout monoliths to form fast flow pathways, such as: macrocracks, separations between grout lifts, and annuli around pipes, supports, and along tank walls. Experiments demonstrated that the size of fast flow pathways that develop, and the peak temperatures attained during hydration, are proportional to the scale of the specimen. Plastic and drying shrinkage commonly led to poor grout-to-metal and grout-to-grout bonding with the capability to transmit fluids, although macroscale flow pathways were not readily observed in bench-scale specimens of cementitious tank grout. Since residual waste is located at the grout-to-metal interface, NRC staff encourages DOE to investigate the plausibility and consequences of such a conceptual model.

<u>Summary of Review</u>

- Due to the overall safety margins in the results analyzed, including uncertainty and sensitivity analyses, NRC finds that DOE has adequately developed appropriate conceptual models and scenarios for the waste residuals at WMA C. For waste evaluations and determinations at other Hanford Site locations, the waste residual and their location may be more risk significant and the issues discussed above would require additional evaluation.

- Future scenario and conceptual model uncertainty are a major source of uncertainty. DOE's safety function methodology is not able to identify 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.

- Recommendations #9 and #10 are discussed this section and included in Table 5-1 of this report.

3.3 Current Climate and Recharge

The following sections provide a summary of DOE's analyses of the current climate and recharge and of the NRC staff's evaluation of DOE's analyses. Current climate and recharge rates provide inputs that can impact rates of material degradation and transport of radionuclides to groundwater.

3.3.1 Summary of DOE Analyses of Current Climate and Recharge

The following sections provide a summary of the information found in the WMA C PA on the current climate and ecology and the recharge rates during operations and post-operations. A portion of the rainwater around the tank farm will infiltrate into the soil; however, vegetation based on the area's climatic condition can reduce the water in the soil through transpiration, thereby playing an important role in the determination of long-term recharge rates to the aquifer.

3.3.1.1 Current Climate and Ecology

In the WMA C PA, DOE provided information on the climatology and ecology of the Hanford Site. Much of this information is obtained from PNNL-15160 (2005) and PNNL-6415 (2007). Typically, the climatological data for the Hanford Site are collected and processed at monitoring sites, and since the early 1980s, key information is transmitted to a meteorology station every 15 minutes. Based on data collected from 1946 through 2004, the average monthly temperatures range 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 of - 0.6 °C (-23.1 °F) was recorded in February 1950. Average annual precipitation at the Hanford Site is 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 occurs during the late autumn and winter. Approximately 50% of total rainfall occurs from November through February with snowfall accounting for about third of that amount. PNNL-15160 (2005) provides the probability of extreme value occurrence, or return periods, from 2 to 1000 years for annual maximum and minimum temperatures, maximum precipitation rates, snowfall amounts, and peak wind gusts.

In PNNL-6415, or the Hanford Site National Environmental Policy Act (NEPA) Characterization, (2007), DOE provided information on the ecology of the Hanford Site and emphasized activities by fauna and flora that may affect exposure pathways and also affect the amount of rainfall that percolates to sufficient depths so as to recharge the saturated zone almost 100 m (328 ft) below. Approximately 6% of the Hanford Site area is covered with buildings such that most of the site is undeveloped. The Hanford Site has been used for agricultural purposes in the past. Past livestock grazing and agricultural tillage precipitated a change in the semi-arid shrub-steppe ecosystem by introducing new non-native species to the Hanford Site that can dominate certain portions of the site (e.g., most of the waste disposal and storage sites are covered by non-native vegetation or are managed to be vegetation-free). Cheat grass, Russian thistle, and knapweed are a few of the invasive species while big sagebrush (*artemisia tridentata*), Sandberg's bluegrass, and bluebrush wheatgrass are native. Typical prominent flora communities in the 200 East Area include both native and non-native species, such as big sagebrush-bunchgrass-cheat grass and cheat grass-bluegrass. The current fauna community includes various bird species, mice species, and relatively larger mammals, such as badgers, coyotes, and mule deer.

Since animal and plant contact with buried waste is a concern, Section 6.3.2.6 in the WMA C PA reported maximum likely root depths and burrowing depths of about 3 m (10 ft) although most are 1.5 m (5 ft) or less. Deeper penetration depths around 200 to 300 cm (6.6 to 9.8 ft) are associated with antelope bitterbrush, big sage, and with insects, specifically harvester ants and solitary bees. Due to the relatively limited data from the Hanford Site, DOE compared this information with other semi-arid sites in the western United States to create a table of maximum penetration depths for biota at the Hanford Site (Table 6-21 in DOE, 2016). A future modified RCRA Subtitle C barrier and the backfill above the tanks will create a thickness of more than 5 m (16.4 ft), thicker than the maximum penetration depths reported in Table 6-21. Consequently, the WMA C PA did not incorporate a biotic pathway within any of its exposure scenarios.

3.3.1.2 Recharge: During Operations and Post-Operations

Groundwater recharge is a major transport process between the wasteform and the aquifer. This section focuses on long-term recharge in the area around WMA C that will not be covered by the engineered surface barrier (Sections 3.4.1.1.2 and 3.4.2.1.2 examine infiltration through the surface cover in more detail). The rate of recharge is determined by natural processes although human activities can influence recharge rates by adding additional water or preventing water from infiltrating into the ground and, therefore, reducing recharge. During the operational period of the Hanford Site, these anthropogenic factors greatly increased the recharge rate to the saturated zone. These factors include water leaks, excavation with water, unsealed abandoned wells, improper drainage control, and other factors. Around the turn of the millennium, anthropogenic recharge has been strongly reduced by decreasing the amount of liquid available at the site. For example, liquids have been removed from the tanks, water pipelines that are no longer needed have been sealed, and surface drainage systems changed. Estimates of water or water and contaminant releases have been made for the WMA C area during operations, but the actual overall amount of additional water added to the system has very large uncertainties. DOE has assumed an average recharge rate of 100 mm/yr (3.9 in/yr) for the operational period which is approximately 29 times larger than the estimated natural recharge rate of 3.5 mm/yr (0.14 in/yr).

Natural recharge occurs when the precipitation (either as rain or snow) that has not been evaporated, transpired, or diverted by surface run-off or subsurface lateral flow has moved sufficiently deep in the unsaturated zone to become deep percolation and, thereafter, groundwater recharge at the saturated zone. DOE stated that large-scale, long-term processes such as climate change, annual precipitation rate variations, and changes in vegetation structure and community are necessary to influence the natural recharge rate for areas with thick unsaturated zones. However, DOE assumed that impacts resulting from plausible climate change would not adversely impact the performance of the surface or unsaturated zone as a barrier, and that episodic precipitation events can be replaced by an average annual infiltration rate because the thickness of the unsaturated zone will dampen the effect of discrete events.

The former assumption is based on the average annual precipitation (200 to 500 mm/yr [7.87 to 19.69 in]) needed for an ecosystem dominated by the big sagebrush (*artemisia tridentata*). This rate of consumption is more than current average annual precipitation rate at the Hanford Site (172.2 mm [6.78 in]), so that the sagebrush community would be capable of exploiting any increases in soil moisture caused by increases in the annual precipitation consistent with, or even in excess, of the previous glacial period.

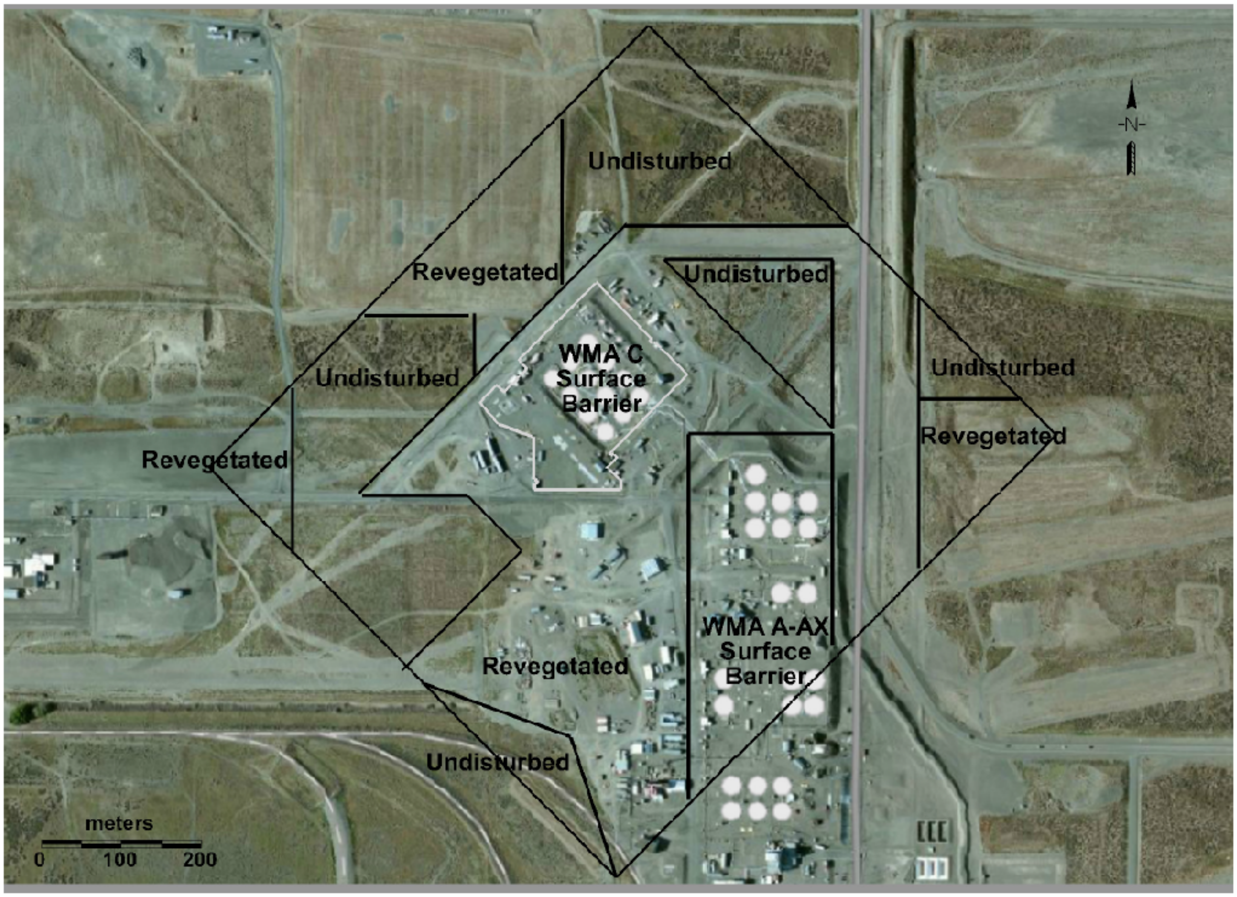
DOE has divided the time period from the immediate past to 10,000 years into the future into subperiods that correspond with different recharge rates based on the surface conditions of the different areas within WMA C and the expected changes to the land cover over time due to change in flora and fauna (see Table 3-2).

Figure 3-3 shows how DOE divided the area around WMA C into different subareas based on the surface conditions (undisturbed and revegetated). The first two subareas are represented with a recharge rate of 3.5 mm/yr (0.14 in/yr), while the surface barrier allows only a rate of 0.5 mm/yr (0.02 in/yr) recharge. Information and assumptions associated with recharge rates in the WMA C PA are based on published data and other reports including PNNL-16688 (2007). PNNL-16688 (2007) included estimated long-term recharge rates for various soil types with shrub covering. Most of these estimates were below the 3.5 mm/yr (0.14 in/yr) used in the WMA C PA. Most of the estimates for soil types without plants were above 20 mm/yr (0.79 in/yr) (e.g., the estimated rate for an unvegetated gravel side slope of a surface barrier was given a 33 mm/yr (1.3 in/yr)). Recommendations in PNNL-16688 (2007) included obtaining more information on the speed and character of restoration of vegetation, particularly with respect to soils that have been extremely disturbed. Specifically, how disturbed and graveled soils allow vegetation to reestablish and how soon they begin to develop the properties close to the original natural soil properties. In addition, recommendations in PNNL-16688 (2007) suggest better understanding of the potential for other processes (e.g., sand dune migration into WMAs) to occur that could affect recharge conditions in the WMAs, including the engineered surface cover performance.

DOE performed a sensitivity analysis and an importance analysis using recharge as one of the parameters. The uncertainty in recharge estimates ranged as high as 5.2 mm/yr (0.2 in/yr) for undisturbed and revegetated disturbed areas and 140 mm/yr (5.5 in/yr) for disturbed areas of WMA C. Based on the barrier importance analysis, long-term recharge (up to 3,400 years after closure) was ranked a distant third in importance (below parameters associated with unsaturated-zone pore-water velocity and saturated-zone flux). Sensitivity analyses involving different combinations of spatial and temporal recharge rates were also performed and shown in the WMA C PA. The recharge rates for WMA C and the area surrounding WMA C during pre-operations (before 1945), operations (1945 to 2020), cover performance (2020 to 2520), and post-cover performance (after 2520) periods were varied and run in five separate sensitivity cases. The maximum concentration at the downgradient point of compliance for the base case was given as 30 pCi/L (1100 Bq/m^3) (Table 8-16 in DOE, 2016). The highest concentration value for any of the five sensitivity runs was 47 pCi/L (1739 Bq/m^3); an approximate fifty percent increase over the base case. Although the increase was relatively modest, the arrival time of the peak concentration was considerably earlier than the base case. This case involved increasing the base-case recharge rate to 140 mm/yr (5.5 in/yr) during operations, back to the assumed rate of 0.5 mm/yr (0.02 in/yr) during the institutional control period, and up to 100 mm/yr (3.9 in/yr) for the post-institutional control period.

Table 3-2 Estimated Pre-Construction, Operational, and Post-Closure Recharge Rates for Various Areas
[WMA C PA (2016), Table 6-6]

Period	**Waste Management Area (WMA) C Region and Surface Condition**	**Base Case Value of Recharge Rate (mm/yr)**
Pre-construction (before 1944)	Undisturbed region (Rupert sand with vegetation)	3.5
Operational period (1945 to 2020)	Undisturbed region (Rupert sand with vegetation)	3.5
	WMA C Surface region (Gravel without vegetation)	100
	WMA A Surface region (Gravel without vegetation)	100
	Disturbed revegetated region (Rupert sand with	22
	Disturbed unvegetated region (Rupert sand with no vegetation)	63
Early post-closure (2020 to 2520)	Undisturbed region (Rupert sand with vegetation)	3.5
	WMA C Surface region (Surface barrier with	0.5
	WMA A Surface region (Surface barrier with vegetation beginning in 2050)	0.5
	Disturbed revegetated region (Rupert sand with vegetation beginning in 2050 with vegetation recovery completed in 2080)	3.5
	Disturbed unvegetated region (Rupert sand with no vegetation until vegetation recovery begins in 2050 and completes in 2080)	3.5
Late post-closure (2520 to 3020 and beyond)	Undisturbed region (Rupert sand with vegetation)	3.5
	WMA C Surface region (Surface barrier with	3.5
	WMA A Surface region (Degraded surface barrier with vegetation begins in 2550)	3.5
	Disturbed revegetated region (Rupert sand with vegetation recovery completed in 2080)	3.5
	Disturbed unvegetated region (Rupert sand with vegetation recovery completed in 2080)	3.5



WMA – Waste Management Area

Figure 3-3 Different Subareas Based on Vegetation and Surface Conditions in the Early Post-closure Time Period
[WMA C PA (2016), Figure 6-17]

3.3.2 NRC Evaluation of Current Climate and Recharge

The following sections discuss the NRC staff's evaluation of the current climate and ecology and the recharge rates during operations and post-operations.

3.3.2.1 NRC Evaluation of Current Climate and Ecology

NRC staff reviewed the information presented in the WMA C PA and supporting documents. Staff determined that information on current climate and ecology and on the exclusion of a biotic pathway from the PA was sufficiently complete and accurate for its intended use. However, many features and processes are dependent on the climatic conditions and interrelationships between precipitation, flora and fauna, infiltration, degradation of features, flow and transport of contaminants and are many and complex. Therefore, NRC staff recommends that for future WIR evaluations and assessments, uncertainty associated with future climates including potential changes to the climate in regions that might affect the WMA C within 1,000 years and within 10,000 years, and the uncertainty in processes that that climate affects (e.g., recharge rates) should be considered in the scenario or conceptual model development in PAs for future WIR evaluations (Recommendation #11).

Current invasive species at the Hanford Site were discussed; however, additional information on the potential transient nature of the mixed natural/non-natural ecosystem should be provided in future PAs. Given the increasing rate of non-native species globally (Early et al., 2016), it is plausible that the current mixture of native and invasive flora and fauna may not be static and that similar ecosystems elsewhere may already be experiencing an influx of newer invasive species. DOE should discuss the full range of uncertainty associated with long-term transient ecosystems at the Hanford Site. Therefore, NRC staff recommends that for future WIR evaluations and assessments, general trends with regards to invasive species in areas that have similar ecosystems as the Hanford Site should be examined, thereby increasing confidence in the long-term recharge rates used in the WMA C PA (Recommendation #12). It was indicated in Gee (1992) that big sagebrush impacted by fires does not come back quickly, which would give the opportunity for invasive species to take over. This is an example of a coupled process that is not captured in DOE's safety function approach to conceptual model development and evaluation.

NRC staff also evaluated impacts of biotic pathways on potential cover designs. Assuming a 4.6 m (15 ft) modified RCRA Subtitle C surface cover will be built over WMA C and a worst-case potential wind erosion rate of 15 cm (6 in) in 500 years (Page 3-118 in WMA C PA), the 4.6 m (15 ft) thickness would be reduced to 1.6 m (4.9 ft) after 10,000 years. Below this thickness lies the interim compacted soil cover of ~0.6 m (2 ft) (Page 6-134 in WMA C PA) and 1 m (3 ft) (Section 6.2.1.1.4 in WMA C PA) of backfill above a typical pipeline. The combined thickness of approximately 3.2 m (10.5 ft) would be thicker than the maximum likely root depths and burrowing depths of about 3 m (10 ft) for biota such that a biotic intrusion into a plugged pipeline would be unlikely. Above the concrete domes of tanks, a backfill thickness of ~1.75 m (5.7 ft) (Fig. 3-45 in WMA C PA) would result in a combined thickness of approximately 3.95 m (13 ft). The top of the 244-CR Vault is above the present-day land surface, but the waste is located more deeply within the structure. Thus, NRC staff agrees that it was not necessary to incorporate a biotic pathway within the exposure scenarios.

3.3.2.2 NRC Evaluation of Recharge: During Operations and Post-Operations

Based on the information presented and reviewed in the WMA C PA and supporting documents, and in conjunction with the performance of the multiple barriers or safety functions, NRC staff determined that DOE has provided sufficient information regarding short- and long-term recharge rates to the groundwater for the long-term continuation of the present climate. However, the uncertainty of potential changes to the climate within 1,000 years and within 10,000 years, and the uncertainty in recharge rates that such changes could create, should be part of the scenario or conceptual model development.

Recharge is usually a sensitive parameter affecting disposal system performance because it relates directly to the rate at which contaminants enter the aquifer. However, with a multi-barrier system, the performance of one barrier may mask the performance of another (i.e., the barriers may be redundant). For example, an effective drainage layer may divert all water away from a deeper protective geomembrane such that a sensitivity case involving the geomembrane may show little effect. The sensitivity and uncertainty analyses in the WMA C PA did not identify recharge as a risk-significant parameter as a result of the modeled performance of the grout and the diffusional release within the grout.

In ORP-63747, Rev. 2 (DOE, 2019) an additional sensitivity case identical to case GRT4 (i.e., grout properties similar to H2 sand) was documented except that the infill grout within the stabilized tank were given hydraulic properties similar to the gravel-dominated backfill. The authors emphasized that the assumption of hydraulic properties similar to either H2 sand or gravel-dominated backfill material was arbitrary and not plausible. The new hydraulic properties of the gravel reduced the contrast between the inside of the tank and the surrounding backfill material to allow advective flow and transport to take place. Peak concentrations increased from 30 pCi/L (1100 Bq/m^3) for the base case to 640 pCi/L ($2.4x10^4$ Bq/m^3) in the new sensitivity case, while the maximum dose was 22 times higher than the base case. Although variations in the recharge rate were not directly analyzed, the new sensitivity case showed the importance of barriers or safety functions that limit the amount of water contacting the waste. One of those barriers which limits the amount of water contacting the waste is the surface cover for which the final design has yet to be determined (see Section 3.4).

The NRC staff determined that DOE has provided sufficient information regarding short- and long-term recharge rates to the groundwater after reviewing information in the WMA C PA and supporting documents. However, the basis for the long-term 3.5 mm/yr (0.14 in/yr) conflicts with other studies and does not include all relevant phenomena. Maher et al. used strontium isotopes studies to derive an infiltration rate of 7 ± 3 mm/yr (0.28 ± 0.12 in/yr) (Maher, 2002). Gee et al. derived an infiltration rate of 14.7 mm/yr (0.58 in/yr) based on evaluation of tritium transport over 16 years (Gee, 1992). These studies would imply that, at a minimum, the uncertainty range prescribed to long-term infiltration rates under non-disturbed conditions should be broader. Therefore, NRC staff recommends that for future WIR evaluations and assessments, DOE should consider increasing the range of recharge for long-term, non-disturbed conditions (Recommendation #13).

The importance of infiltration and of recharge rates on the arrival time of the peak concentration at the point of calculation is demonstrated by the sensitivity case INF03 documented in Section 8.2.1 of the WMA C PA. For this sensitivity case, base case conditions are unchanged, except for a net infiltration rate of 140 mm/yr (5.5 in/yr) during the operational period, surface cover performance shortened to last as long as the institutional control period (i.e., 100 years after closure), and a post-institutional control net infiltration rate of 100 mm/yr (4 in/yr). Peak concentration of ^{99}Tc is approximately 50% higher in INF03 than that of the base case; however, the arrival time after the surface cover stops performing, in this case 100 years, is more than an order of magnitude shorter than the base case arrival time, 74 years compared to 1055 years. An annual average recharge rate of 100 mm/yr (4 in/yr) is comparable to the average recharge in Iowa or Illinois (Reitz et al., 2017) and, therefore, not extreme. The uncertainty of potential changes to the climate or land use within 1,000 years and within 10,000 years would need to be part of a scenario development. The uncertainty in future infiltration rates and subsequently recharge rates would not only have an effect the travel time of radionuclides within the unsaturated zone but on numerous FEPs that may drive contaminant concentrations down (e.g., greater aquifer thickness).

Recharge at Hanford, as shown in photographs of present-day conditions, is likely to be episodic. In 1985, a strong, warm wind melted approximately 200 mm (8 in) of snow in one day

(Gee, 1992). Studies to develop long-term values must be appropriately broad in scope to capture the important spatial and temporal phenomena. The profile and conditions at the land surface are likely to dictate the long-term infiltration rates with one of the determining factors being the relative portion of sand compared to fine-grained sediments (silt). Numerous sand dunes are located currently in and around the Hanford Site. If the final closure cover has more than very minimal slopes, sand is likely to be deposited on portions of the cover from natural aeolian processes.

DOE should consider additional research to address how moderately to extremely disturbed areas would allow vegetation to become fully reestablished and how soon the soils begin to develop normal soil properties such that they start to resemble natural soils. While more data on revegetation for moderately disturbed soil would provide support for assumptions concerning the speed and character of restoration, this need is particularly true for soils that have been extremely disturbed (PNNL-16688). Table 6-6 in the WMA C PA shows a recharge rate of 63 mm/yr (2.5 in/yr) for sand with no vegetation during the operational period, which is a much higher rate than the natural recharge rate (3.5 mm/yr [0.14 in/yr]) assigned for the disturbed revegetated regions during the post-closure. DOE considers natural disturbance processes such as replacement of vegetation with invasive species or brush fires to be unlikely or short-term. Considering the potential impacts on recharge rates, additional basis would provide more confidence in DOE's long-term recharge rates. Therefore, NRC staff recommends that for future WIR evaluations and assessments, additional information is needed on the speed and character of revegetating a disturbed area, particularly extremely disturbed areas, and if revegetated areas have the same recharge rate as undisturbed areas with natural soil properties (Recommendation #14).

DOE claimed that a sagebrush-dominated ecosystem would be capable of exploiting any increases in soil moisture caused by increases in the annual precipitation consistent with or even in excess of the previous glacial period. However, what is not clear is if the long-term dominant fauna at WMA C will be a big sagebrush community. Sagebrush does not re-sprout after fires (Gee, 1992). In Section 3.1.3, the WMA C PA listed big sagebrush as an example of a fire-intolerant species and, in addition, the precipitation rate demand associated with a sagebrush-dominated ecosystem (200 to 500 mm/yr [7.87 to 19.69 in]) is already above the present average annual precipitation (172.2 mm [6.78 in]) at the Hanford Site. A future drier climate with an increasing frequency of fires may hinder sagebrush-dominated ecosystems from forming. If institutional controls no longer apply, livestock grazing, agricultural production, and range fires may contribute to colonization and dominance by non-sagebrush vegetation species. Since DOE is relying on plant communities such as the sagebrush community to exploit any increases in soil moisture caused by increases in the annual precipitation consistent with, or even in excess, of the previous glacial period, NRC staff recommends that for future WIR evaluations and assessments, DOE should reevaluate the ability of the ecosystem to prevent increased recharge if the ecosystem changed or if soil moisture increases. Specifically, DOE should evaluate a transient ecosystem at the Hanford Site where big sagebrush is not the dominant fauna (Recommendation #15).

Summary of Review

- NRC staff reviewed the information presented in the WMA C PA and supporting documents and determined that information on the current climate and its associated ecology, and on the exclusion of a biotic pathway from the PA, was sufficiently complete and accurate for its intended use.

- Based on the information presented and reviewed in the WMA C PA and supporting documents, and in conjunction with the performance of the multiple barriers or safety functions, NRC staff determined that DOE has provided sufficient information regarding recharge rates to the groundwater based on a long-term continuation of the present climate.

- Many features and process are dependent on the climatic conditions. Interrelationships and interdependencies between, for example, perception, flora and fauna, infiltration, barrier degradation, waste release, flow and transport of contaminants, and recharge are many, complex, and potentially increase uncertainty.

- Recommendations #11 through #15 are discussed this section and included in Table 5-1 of this report.

3.4 Engineered Barrier System

This section contains the summary of DOE analyses of the engineered barrier system and the NRC evaluation of the engineered barrier system.

3.4.1 Summary of DOE Analyses of Engineered Barrier System

In the WMA C PA, DOE modeled 19 separate facilities (22 facilities if the four 244-CR vault tanks are considered separately) that contain radioactive waste and may release contaminants in the future. The single-shell tanks make up 16 of the 19 facilities, while ancillary equipment make up the others (see Section 1.3). The planned engineered barrier system for WMA C is intended to delay and slow the release rate of the contaminants and will consist of an engineered surface cover and subsurface barriers. The subsurface barriers include both reinforced concrete (i.e., concrete with rebar) surrounding 18 out of the 19 separate facilities with cementitious grout within these facilities. In addition, the wasteforms themselves can be a barrier to the release of radioactivity since the residual waste is conceptualized to be sludge-like with a texture similar to a hardened paleosol (Page 6-10 in WMA C PA).

Some of the subsurface barriers have been in place since the tank farm was constructed. These include the existing reinforced concrete vaults that surround the 16 single-shell tanks, the C-301 catch tank, and the 244-CR vault tanks. The four valve pit/boxes and the seven diversion boxes are also made of concrete; however, the WMA C PA assumes that the only waste remaining in these structures at closure will be limited to waste adsorbed to the concrete and that the relatively small inventory can be incorporated as part of the pipeline source term (Page 6-9 in WMA C PA). The pipeline source term is simulated as a single source area reflective of the approximate areal distribution of the waste transfer pipelines. Instead of modeling discrete source terms for the approximately 11.2 km (7 mi) of discrete pipelines, the single source term used for the pipelines was located over WMA C and represented by a volume with side dimensions of 150 m (492 ft) and a depth of 0.076 m (0.25 ft) (see black-lined square in Figure 3-4). Pipeline walls were assumed to be absent, and the inside of the pipes was assumed to not be grouted. The pipelines were assumed to possess no structural integrity such that both advective and diffusive releases occur after closure. Hence, the pipeline source term is the only facility out of the 19 separate facilities assessed that was not associated with a subsurface concrete vault barrier.

In Appendix H of the WMA C PA (see Table H-1), DOE specifically identified safety functions associated with the engineered barrier system that provide specific functions and are relevant to the performance of the facility. This performance may be related to the function's ability to slow or prevent infiltration or human intrusion, slow release of contaminants through hydraulic or chemical properties, or provide structural stability via its physical properties. This section and the following section will discuss and evaluate performance of the following safety functions listed in DOE's WMA C PA (2016): EB1 - RCRA Cover (infiltration reduction); EB2 RCRA Cover (depth of disposal); EB8 - Tank structure (permeability); EB9 - Grout in tank (permeability); and EB13 - Tank Base Mat (permeability).

DOE discussed the conceptual model whereby these safety functions slow and delay the release of contaminants in the WMA C PA (DOE, 2016). The WMA C PA model domain includes the actual WMA C and 100s of meters of surrounding area. Most of the WMA C PA model domain is assumed to have a long-term, post-closure infiltration rate of 3.5 mm/yr (0.14 in/yr), which corresponds to the infiltration rate in an undisturbed area with native vegetation. In the WMA C PA, DOE assumed a modified RCRA Subtitle C cover overlies the actual WMA C although specific design information is not yet known. While the future surface barrier may potentially last longer or shorter than the design life, the WMA C PA assumed a surface barrier with an infiltration rate of 0.5 mm/yr (0.019 in/yr]) up to 500 years after closure and 3.5 mm/yr (0.14 in/yr) thereafter. The assumed surface cover outline is visible in blue in Figure 3-4. The pipeline area to the southwest of the CR diversion boxes and CR-Vault was not simulated as being covered in the STOMP© model (see Figure 7-22 in the WMA C PA).

Although there is substantial uncertainty about how long the subsurface tank wall may last before physical and chemical degradation occurs and allows water to flow through it, based on evaluations of the durability of the material and the longevity of its function to reduce water flow, the tank structure and infill grout together were assumed by DOE to provide a low-permeability barrier to infiltrating water flow for thousands of years. Water that does infiltrate through the engineered surface cover will subsequently be impeded from contacting the residual waste by the engineered subsurface barriers including the reinforced concrete dome, sidewall, and floor and basemat of the tank shell, in addition to the infill grout and waste. DOE assumed all cementitious material was saturated. The only transport mechanism for contaminant release from the residual tank waste was by diffusion through the cementitious material and into the unsaturated zone. The presence of continuous water connections was also assumed across the grout and concrete layers for the diffusive transport to occur in the aqueous phase.

Matrix-degradation-rate-based release (e.g., ^{99}Tc) and solubility-controlled release (e.g., uranium) control the dissolved concentration of contaminants in the residual waste pore volume (see Section 3.5). The transfer pipeline source term was assumed not to have the same subsurface barriers as the other facilities and the source release model consisted of waste uniformly spread over the area with both advective and diffusive releases occurring. Consequently, the first radionuclides to arrive 100 m (328 ft) downgradient of the WMA C perimeter were from the pipeline source term.

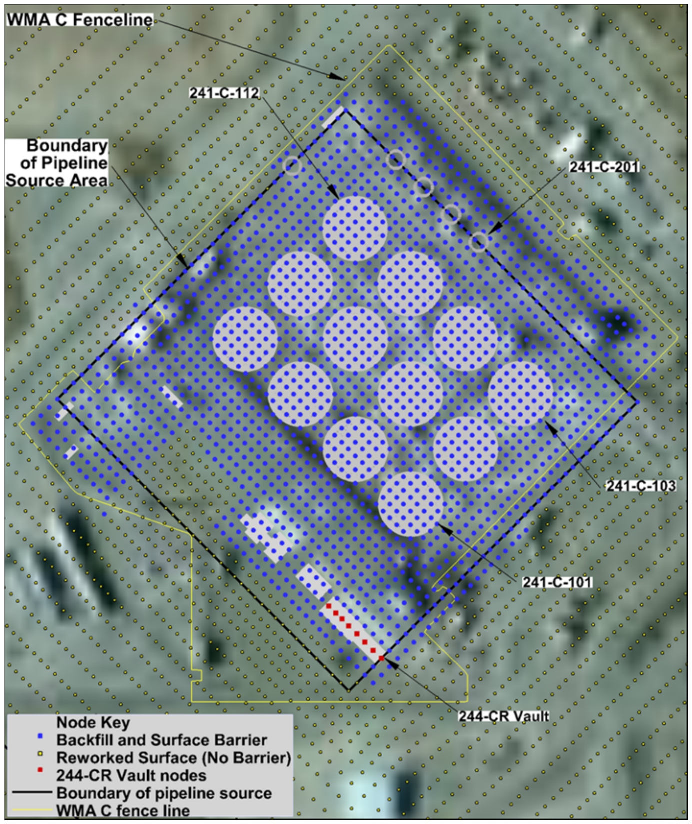
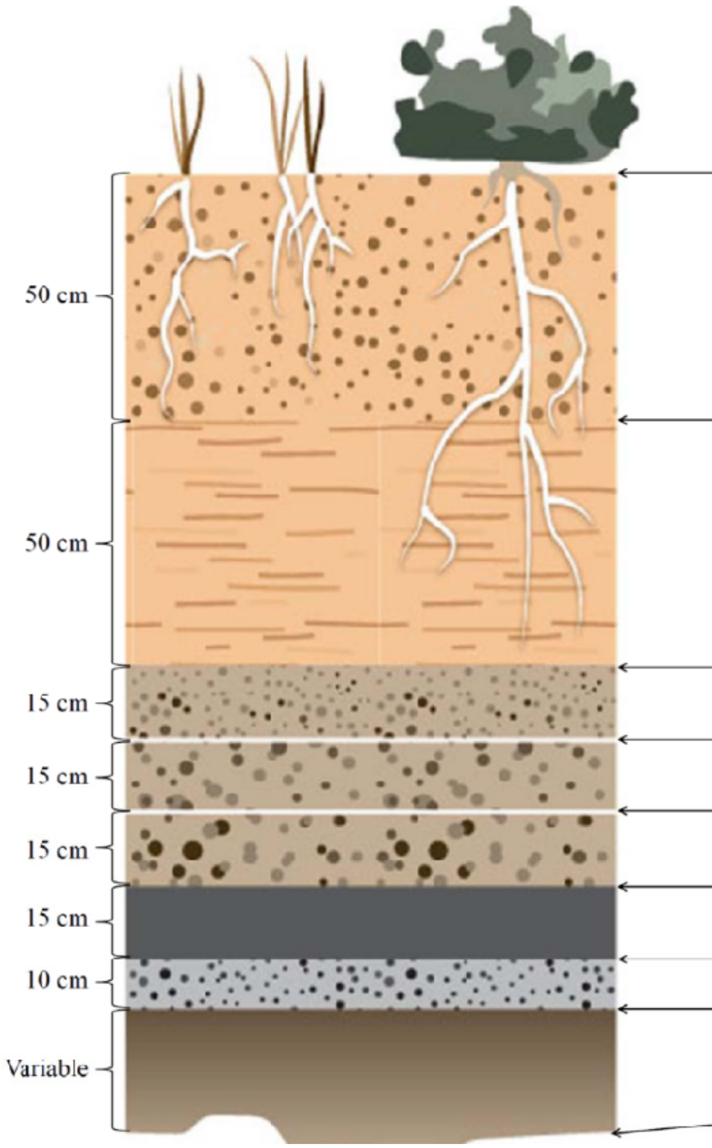


Figure 3-4 Layout of Pipeline Source and Surface Cover to be Used in the WMA C Three-Dimensional Model Domain
[NRC, 2018b]

3.4.1.1 Engineered Surface Cover

The single-shell tanks, tanks in the 244-CR vaults, and the C-301 catch tank will be filled with a cementitious grout from the tank bottom up to the overlying tank roofs which will lie beneath backfill with a thickness of ~1.75 m (5.7 ft) and an interim ~0.6 m (2 ft) compacted soil cover resulting in a combined thickness of approximately 3.85 m (12.6 ft). Residual waste at the bottom of the tank will be encapsulated by this grout. Once grouting has been completed, a multilayer engineered surface cover and drainage system will be constructed on top of this combined layer.

Although DOE has not committed to a specific cover design, the current closure plan approach would be to place an engineered modified RCRA Subtitle C barrier over WMA C (Section 3.2.1.2.2 in WMA C PA). Figure 3-5 shows a cross section of this planned engineered surface cover. The cover is designed to: 1) limit infiltration by promoting runoff, evapotranspiration, and the shedding of water away from the tanks and ancillary equipment; 2) provide physical stabilization of the site; and 3) act as an intruder deterrent.



Layer Number	Thickness cm (in)	Layer Description
0	N/A	Cover Vegetation
1	50 (20)	Silt loam with pea gravel admix
2	50 (20)	Compacted topsoil
3	15 (6)	Sand Filter
4	15 (6)	Gravel Filter
5	15 (6)	Lateral drainage aggregate
6	15 (6)	Asphaltic concrete with spray applied asphalt coating
7	10 (4)	Asphalt base course
8	Variable	Grading Fill

Figure 3-5 Modified RCRA Subtitle C Barrier Profile
[Figure 3-2 in DOE, 1996]

After installation of the surface barrier and closure, a 100-yr active institutional control period will begin, during which active maintenance will be conducted and significant erosion repaired. The modified RCRA Subtitle C barrier generally consists of a layer of clay, geomembrane material, a layer of asphaltic material, and sand and gravel with a combined thickness of 1.7 m (5.6 ft). Below these layers, additional grading fill will be placed so that the cover will have a total thickness of 4.6 m (15 ft).

The vegetated surface layer of fine-grained soils, designed to retain moisture and encourage evapotranspiration, may be increased in thickness to provide additional defense-in-depth against direct contact exposure from a potential future basement excavation. DOE stated the performance will be dependent on the various engineered layers shown in Figure 3-5 and the expected performance is based 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 a WMA C cover design similar to the Prototype Hanford Barrier could be sufficiently robust and perform as designed for the long-term. In Section 3.2.1.2.2 in the WMA C PA (DOE, 2016), DOE stated that the Prototype Hanford Barrier is expected to continue to perform even after fires have burned off the vegetation (PNNL-18934, 2009) and extreme precipitation events (i.e., 70 mm (2.8 in) in 8 hours representing a 1,000-year return storm for Hanford) (PNNL-14143, 2002). For purposes of slowing the infiltration rate, DOE indicated it is the upper layer of silt loam that is expected to provide much of the performance by storing precipitation that falls over the site during the winter months and removing excess moisture by evaporation and transpiration during the summer months. Despite natural failure mechanisms such as bioturbation of the silt loam layer, wind erosion, or accretion of windblown sand, DOE stated in PNNL-14744 (2004) that with appropriate design considerations, the long-term effectiveness of the surface barrier would continue to limit recharge rates to less than 0.1 mm/yr (0.004 in) for thousands of years though DOE did not assign that level of performance in the analyses.

3.4.1.1.1 Erosion

The rate of erosion must be kept low to maintain the engineered surface cover's ability to reduce infiltration, to provide stability for the disposal facility for long time periods, and to deter human intrusion. Currently, the planned thickness of the cover is more than 3 m (10 ft) thick, the nominal depth typically considered for human excavation activities.

Water and wind erosion of the surface cover material can impact the integrity of a surface cover, the vegetation, and upper soil loam layer, which provide water storage and promote evapotranspiration. The Prototype Barrier Treatability Test Report, DOE/RL-99-11 (DOE, 1999b) evaluated the potential for wind erosion for surface barriers and DOE reported that an erosion rate of 15 cm (6 in) of silt loam in 500 years was possible. In the WMA C PA, DOE pointed out that the modified RCRA-compliant closure cover will not be derived from agricultural soils as investigated in DOE/RL-99-11 (DOE, 1999b); instead, the intended cover will have a mixture of fine-grained soil and pea gravel on top to significantly reduce wind erosion. Based on PNL-8478 (1993) and WHC-EP-0673 (1994), a wind erosion reduction of 96% was expected, such that wind erosion rate would decrease to 15 cm (6 in.) in 12,500 years. In addition, the WMA C PA stated that wind erosion of the silt loam for a vegetated surface barrier should be minor, based on the experience at the Prototype Hanford Barrier.

One of the phenomena most likely to affect long-term stability is surface water erosion. DOE stated in Section 3.2.1.2.2 of the WMA C PA that "low precipitation, the low intensity of precipitation events, the absence of surface run-on features at the Hanford Site, and stability monitoring (PNNL-18845, 2011) all support the assumption that water erosion will not be a significant factor at WMA C barrier."

3.4.1.1.2 Infiltration

DOE stated that the engineered surface barrier is expected to perform like a modified RCRA Subtitle C barrier (Section 3.2.1.2.2 in WMA C PA), which in turn should function similarly to the

Prototype Hanford Barrier (PNNL-16688, 2007). DOE believed data from the Prototype Hanford Barrier and other surface barriers indicate that the barrier will be capable of limiting recharge to less than 0.1 mm/yr (0.004 in/yr) even with a complete lack of vegetation (i.e., only evaporation and no transpiration). In addition, according to PNNL-13033 (1999), not even the erosion of the silt loam layer and deposition of dune sand on the barrier is likely to significantly alter the performance of the barrier. In the WMA C PA, DOE used a recharge rate of five times that rate (i.e., 0.5 mm/yr (0.02 in/yr)) and, therefore, was consistent with the drainage design specification in DOE/RL-93-33 (DOE, 1996). No gradual degradation was modeled for the surface cover. The degradation of the surface cover occurred at 500 years after closure and the infiltration rate was assumed to increase to 3.5 mm/yr (0.14 in/yr) for the remainder of the simulation. The value of 3.5 mm/yr (0.14 in/yr) was also used for the area surrounding WMA C and corresponded to the recharge in an undisturbed area where native vegetation is assumed to reclaim the land. The side slopes and berm were included as part of the barrier surface and their impact on the overall recharge rate was expected to be relatively negligible. In PNNL-16688 (2007) DOE indicated that the long-term recharge rate for the sandy gravel/gravelly sand features will be 1.9 mm/yr (0.07 in/yr), which is comparable to the Burbank loamy sand they are expected to evolve into, and less than the 3.5 mm/yr (0.14 in/yr) used in the analysis for the degraded barrier surface. The engineered surface cover and processes associated with this feature were not modeled directly in the WMA C PA, so that the recommended net infiltration rates were applied in a spatially and temporally appropriate manner to the area under the engineered cover.

Limiting infiltration reduces the potential for advective transport of radionuclides. Although processes within the engineered surface cover were not modeled (e.g., evaporation, transpiration, lateral drainage), sensitivity and uncertainty cases were run (Sections 8.2.1 and 8.1.3.1, respectively, in the WMA C PA) in order to address potential increased infiltration/recharge that may occur as a result of a variety of long-term changes in the FEPs. A variety of sensitivity cases were run involving the surface barrier flow safety function. However, the safety functions associated with the engineered subsurface barriers dominate performance and none of the sensitivity cases involving infiltration rates had much effect on radionuclide concentrations. Increased infiltration through the surface barrier after closure was unable to reach the residual waste in the tanks. Infiltrating water was diverted around the grout-filled concrete vaults containing the waste. Sensitivity case INF03 resulted in the greatest output change and involved a degraded surface barrier infiltration rate increase from 3.5 mm/yr (0.14 in/yr) to 100 mm (3.94 in/yr), or a factor of 29 increase. Sensitivity case INF03 shortened the arrival time of the maximum concentration at the point of calculation (POC) such that the peak arrived over 1,400 years earlier; however, the concentration itself only increased from 30 pCi/L (1100 Bq/m^3) for the base case to 47 pCi/L (1740 Bq/m^3) for case INF03. A sensitivity case with no engineered surface barrier (i.e., a one-off analysis) was not run.

The uncertainty in recharge rates were considered, and Table 3-3 reproduces the spatial and temporal range of recharge rates considered for the WMA C PA. Based on the uncertainty analysis results, parameters important to the groundwater pathway were associated with a late post-closure degraded surface barrier and higher infiltration rates (Table 8-3 in WMA C PA, 2016). Sensitivity runs showed similar results.

Table 3-3 Spatial and Temporal Uncertainty in Recharge Rates Considered for WMA C [Table 8-3 in DOE, 2016]

Spatially Distinct Zone	Uncertainty in Recharge Estimate (mm/yr)		
	Minimum	Maximum	Most Likely
Pre-operations Period (Prior to Year 1945)			
Undisturbed (Natural Vegetation)[a]	0.5	5.2	1.9
Operations Period (Years 1945 – 2020)			
Undisturbed (Natural Vegetation)[a]	0.5	5.2	1.9
Waste Management Area C Disturbed Surface	40	140	100
Non-Tank Farm Disturbed Surface[b]	22 / 63	22 / 63	—
Post-Closure Period with Intact Surface Cover (Years 2020 – 2520)			
Undisturbed (Natural Vegetation)[a]	0.5	5.2	1.9
Waste Management Area C Surface Barrier	0.1	0.9	0.5
Non-Tank Farm Disturbed Surface[a,c]	0.5	5.2	1.9
Late Post-Closure Period with Degraded Surface Cover (Years > 2520)			
Undisturbed (Natural Vegetation)[a]	0.5	5.2	1.9
Waste Management Area C Surface Barrier	0.5	5.2	1.9
Non-Tank Farm Disturbed Surface[a]	0.5	5.2	1.9

[a]Waste Management Area C late post-closure surface barrier assumed to acquire natural vegetation recharge rate.
[b]Disturbed areas that allow vegetation are assigned 22 mm/yr. Disturbed areas that are reworked such that vegetation does not grow are assigned 63 mm/yr.

3.4.1.2 Engineered Subsurface Barrier

The following sections provide a summary of the information found in the WMA C PA on the degradation of the pre-closure cementitious material and steel and of the infill grout and wasteform.

3.4.1.2.1 Degradation of Pre-Closure Cementitious Material and Steel

In the report RPP-RPT-46879, Rev. 3 (2016) DOE assessed the potential for various degradation mechanisms to impact steel, concrete, and grout currently found at WMA C or included in the current designs. The degradation mechanisms evaluated in this report included the effects of concrete curing, elevated temperatures, radiation, freezing and thawing, and creep and shrinkage, in addition to the reaction of aggregates and alkalis and the leaching of calcium hydroxide, also known as carbonation. In RPP-RPT-46879 (2016), DOE evaluated the structural integrity of existing single-shell tanks and other features at the Hanford Site. This allowed data to be collected on the reinforced steel (or rebar in the tank structures). Various structures components were inspected (e.g., tank domes and sidewalls of existing tanks) including by remote visual inspections.

A 1.4-m (55-in) diameter concrete plug or section was removed from the center of the dome of Tank C-107. No cracks or large air voids were found during the inspection process, and the protective asphaltic membrane and mortar layers near the top of the dome were found to be intact. Petrographic examination results showed the concrete to be in good condition. Macroscopically, the cores contained no large voids and the cement was nearly completely hydrated. The depth of carbonation from the top surface of both cores was reported to be 1 to 2 mm (0.04 to 0.08 in). A core sample from the sidewall of Tank A-106 was obtained in May 2014 with a total length of over 11.6 m (38 ft) and extending down to approximately halfway through the tank footing (RPP-RPT-46879, 2016). This tank was chosen due to its high heat load history and concerns over the thermal degradation of the concrete from heat exposure. The results, however, showed that the effects of thermal degradation on the mechanical properties of the Tank A-106 concrete appeared to be negligible. One crack and a few microcracks were observed in the examined core segments. Some amount of carbonation was observed in the paste, along with a small amount of secondary ettringite mineral, possibly from sulfate containing impurities in the paste. The depth of carbonation was found to be shallow and about 1 to 4 mm (0.4 to 0.2 in) from the outer surface in some core segments.

Various core samples were taken from Tank SX-115 (37 samples) and the 202-A PUREX Canyon Building (17 samples). Although most cores showed no visible signs of concrete deterioration, four tank farm cores and two PUREX Building cores were found to have visible cracks, ranging from ~5 to 25 cm (2 to 10 in). The report was unclear if the cracks had been caused by the actual coring activities. However, air voids up to 2.4 cm (1 in) long were also detected and these are likely indicators of poor concrete placement during construction. Compression strength and tensile strength tests on Tank SX-115 concrete samples taken in year 1981 were above the design value but indicated a pattern of possible decreasing strength with depth. A comparison between samples from the A-Tank Farm and SX-Tank Farm showed that the A-Tank Farm concrete compressive and tensile strengths were about 45 percent greater than those values obtained from Tank SX-115 concrete; however, the elastic properties were essentially equal. In RPP-RPT-46879 (2016) DOE offered possible reasons for the differences including variations between concrete batches or that SX-115 tank wall concrete was subjected to higher temperatures during tank farm operations than the dome concrete in the A-Tank Farm.

Table 6-2 in the WMA C PA showed DOE's calculated bounding depth due to carbonation for buried concrete for different exposure times. It was assumed that the maximum depth of carbonation within the buried concrete tank structure was 10 mm (0.39 in) and not the 1 to 4 mm (0.4 to 0.2 in) carbonation depth measured on Tank A-106 core segments. A simple analytical expression was used to calculate the depth of degradation: $x_c = At^{1/2}$, where "x_c" is the depth of the carbonation front, "A" is a proportionality constant linked to carbon dioxide, and "t" is exposure time. Using 70 years as the exposure time and a calculated value of 1.2 mm (0.05 in) for the constant A, the non-linear movement of the carbonation front was calculated to reach a depth of 84.5 mm (3.3 in) 5,000 years after closure.

In RPP-RPT-46879 (2016) DOE evaluated steel corrosion and degradation including the potential corrosion of the transfer pipelines and of the tank liners. There is a large uncertainty as to the integrity of the pipelines due to the lack of specific data and information, and the numerous degradation and failure mechanisms for pipelines through thermal, fatigue/wear, chemical corrosion welds, and gasket failures. Therefore, DOE did not include pipeline features in the PA calculations and mass release occurred from advection and diffusion. Similarly, for the liners of the single-shell tanks, no waste isolating capabilities were assumed for the tank liner and no credit taken in the PA calculations. Fill and empty cycles and temperature changes due to thermal and operational loads have led to fatigue in the steel liners and piping. In combination with hot and corrosive waste that were sometimes stored in the tanks in addition to treatment having been performed to remove stresses in the weldments, DOE decided that the uncertainty in the tank liner integrity is too high for credit to be taken (see Section 6.2.1.1.1 in the WMA C PA).

In RPP-RPT-46879 (2016,) DOE discussed rebar corrosion in the presence of Cl^- or CO_2. A high pH value of the concrete should provide steel an adherent and protective magnetite passive film that inhibits corrosion especially pit corrosion. Lowering the pH near the rebar such that the protective magnetite layer is not stable and may accelerate the corrosion rate of rebar in the passive state. However, DOE concluded that chloride concentrations are low in the wastes and groundwater, and thus rebar corrosion is not considered a significant issue. DOE stated that, for the base case, rebar is assumed to stay intact for at least 20,000 years (NRC, 2018c).

In the WMA C PA, DOE discussed aboveground concrete structures at the Hanford Site and other natural analogues. A minor amount of microcracking and ettringite was observed in the aboveground structures and an upper bound carbonation rate for the Hanford Site was obtained based on the few specimens; the carbonation rate was calculated to be about 0.3 to 0.9 mm/yr (0.01 to 0.04 in). The natural analogues selected included ancient structures such as the Roman aqueducts, the Pantheon and Hadrian's Wall. Natural analogues that included concrete structures with reinforced steel were not included since this combination of materials was not widely applied until the end of the nineteenth century.

In RPP-RPT-46879 (2016), DOE determined that the most likely degradation scenarios for the systems, structures, and components considered involved localized areas of structural degradation due to carbonation and reactions of aggregates and alkalis caused by water infiltration and that, with the exception of extreme elevated temperatures, the other degradation mechanisms for cementitious materials would not significantly contribute to concrete degradation at WMA C. In RPP-10435 (2002), DOE stated that the primary potential degradation mechanisms for the single-shelled tanks are corrosion of the reinforcing bars, degradation of the concrete mechanical properties due to past high temperature exposure, and caustic waste chemical exposure damage of the concrete in leaking tanks. DOE further stated that the most significant structural uncertainty is the condition of the reinforced concrete basemat and footing, due to the inaccessibility for inspection.

Based on the tank infill grout and concrete degradation mechanisms discussed in RPP-RPT-46879 (2016) and Section 6.2.1.2 of the WMA C PA, DOE determined that tanks are not likely to be fully-degraded within the modeled time period of 10,000 years (Page 6-66 in WMA C PA). The primary contaminant transport process in the PA is diffusion with a negligible amount of advection. The concentration differences across the grout and concrete tank basemat provide a gradient leading to diffusion. The importance of advective flow was demonstrated by DOE in their response (DOE, 2019) to the NRC staff's RAI 2-9 (NRC, 2019) when DOE modified the original sensitivity case GRT4 of the concrete shell and infill grout properties from Hanford H2 sand values to hydraulic property values similar to those of the surrounding gravel-dominated backfill material while holding the infiltration rate constant. DOE emphasized that there are no known FEPs or combination of FEPs that could produce this condition and that the analysis is intended to evaluate the loss of diffusion-only release. This modified sensitivity case increased the maximum concentration at the downgradient POC for the base case value from 1,100 Bq/m^3 (30 pCi/L) to $2.4x10^4$ Bq/m^3 (640 pCi/L), or an increase by a factor of 22. Degradation of the concrete shell and infill grout allows other transport mechanisms besides diffusion to take place and, thereby, significantly changes the contaminant concentration and the dose, in this case from $1x10^{-3}$ mSv/yr (0.1 mrem/yr) to 0.022 mSv/yr (2.2 mrem/yr).

3.4.1.2.2 Degradation of Infill Grout and Wasteform

To provide for mechanical stability and to decrease the release of residual waste from closed systems, DOE plans to fill tanks and ancillary equipment (besides pipelines) with grout. The infill grout is assumed to provide an impermeable barrier to flow, based on DOE experience with grouting of other waste tanks (SRNL-STI-2012-00578). At the time of the development of the Draft WIR Evaluation for WMA C, the formulation of the grout and the testing to determine that the grout would achieve the design objectives had not yet been completed. In the base case assessment, it was assumed that no advective release of radioactivity occurs from the residual waste.

There is not an engineered wasteform, per se, in the PA model. The residual waste is represented as a thin layer spread uniformly over the plan view area (looking down from the top) of the structure being simulated (e.g., 100-series tank). The pore space of the waste layer is assumed to be 100% saturated with water and the porosity was assigned a value of 40% based on evaluation of sludge retrieved from tanks. The release of two key radionuclides, ^{99}Tc and U, was based on empirical measurements from actual tank waste (PNNL-20616). Otherwise, the wasteform part of the model does not degrade or otherwise alter its properties in the simulation. The infill grout is assumed to control the chemical conditions of the pore water that contacts the residual waste. Waste is available to diffuse to a concrete basemat layer underneath the tank. The effect of steel liner of the tank is not included in the calculations. DOE stated that ongoing chemical and physical degradation of the tank wall concrete and grout leading to the eventual formation of cracks (which could then lead to advective flow) was included in the conceptual model. However, these processes were not included in base case simulations because the rates of the processes were not judged to be sufficient to alter the properties of the materials by a significant amount. The base of the tank systems is reinforced concrete at least 0.15 m (6 in) thick with an additional 0.05 m (2 in) of grout on top of the reinforced concrete layer.

3.4.2 NRC Evaluation of Engineered Barrier System

The engineered barriers the NRC staff evaluated include the engineered surface cover and the engineered subsurface barriers. The engineered subsurface barriers include the reinforced concrete structures and the infill tank grout.

3.4.2.1 NRC Evaluation of Engineered Surface Cover

The engineered surface cover design information that DOE provided to the NRC staff is preliminary and will be finalized at a later date. As previously stated, the cover is designed to 1) limit infiltration, 2) provide physical stabilization of the site, and 3) act as an intruder deterrent. Although the design for long-term erosion control and long-term stability are rather similar, cover attributes designed for one function (e.g., rip rap decreases the erosion rate) may not be beneficial for another function (e.g., rip rap increases the infiltration rate).

3.4.2.1.1 NRC Evaluation of Erosion

Sustaining a thick layer of soil on an engineered surface cover without active maintenance is desirable since this will prevent or reduce the impact of intrusion into the waste and provide physical stability for the site. In order to maintain the thickness of this layer, erosion must be kept at a minimum and controls must be in place for extreme precipitation events. As previously stated, in Section 3.2.1.2.2 of the WMA C PA, DOE discussed low precipitation, the low intensity of precipitation events, and the absence of surface run-on features at the Hanford Site that would result from these events as a basis as to why water erosion will not be a significant process affecting the surface cover. Despite the extensive reference material associated with the Hanford Protype Barrier, NRC staff were not able to identify the technical basis supporting the absence of surface run-on features.

Field studies documented in PNNL-14143 (2002) involved one-half of the soil surface being irrigated for three years such that simulated precipitation averaged three times the long-term annual average (480 mm/yr or 19 in/yr). More intense rainfall representing a 1,000-year return storm for the Hanford Site was applied in March for three years (70 millimeters or 2.8 inches in 8 hours). After the first year of testing, there was no measurable wind erosion, and runoff was small and mainly associated with rapid snowmelt. The side slopes and soil cover remained stable. Limited erosion was attributed to extensive revegetation of the soil surface. Although the information from these field studies provide useful information, erosion protection designs must be based on an appropriately conservative rainfall event and that more rigorous analyses are required that include determining the Probable Maximum Precipitation (PMP) and designing a surface barrier for such an event. The PMP is defined as the theoretically greatest depth of precipitation that is possible during a given time period over a given area at a geographic location. The design criteria for the main component of the cover, the side slopes, and the toe of the side slopes should be similar to the methodologies and approaches found in the NRC's NUREG-1623 (2002), which addresses a 1,000-year timeframe. Therefore, the NRC staff recommends that for the final Waste Evaluation for WMA C, 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 (Recommendation #16).

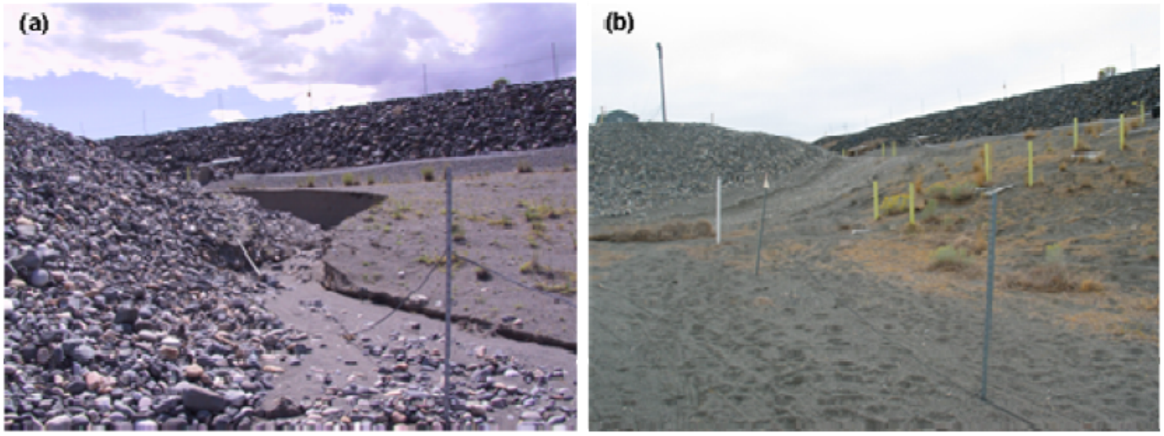


Figure 3-6 Erosion at the Hanford Barrier from a Severe Thunderstorm

Water erosion has occurred on the Hanford Prototype Barrier as shown in photos from Section 2.2.4 in PNNL-17176 (2007) (shown in Figure 3-6). Severe thunderstorms with sufficient intensity to allow water to collect near the BY Farm eroded a berm, flowed down the northwestern slope of that tank farm, and eroded gravel armor and created a channel over 1 meter (40 inches) deep at the base of the barrier side slope. DOE has emphasized that the erosion was due to runoff from the BY-Tank Farm and not from the cover itself (NRC, 2018d). This episode, however, emphasizes the need for an engineered cover design in line with a PMP analysis. Alternatively, less performance credit for the engineered cover could be prescribed consistent with the expected rates of erosion of a less robust engineered cover design.

At the Hanford Site, wind erosion is considered more of a concern than water erosion; DOE presented information on this process. The cover may have a mixture of fine-grained soil and pea gravel to reduce wind erosion. In the WMA C PA, DOE claimed this mixture should reduce wind erosion considerably. Numerous developers of engineered covers have been actively considering using this mixture in the top layer to reduce erosion, and literature on this topic supports such claims (Li et al., 2001; Waugh, 2004).

Regarding the requirements of the engineered surface barrier associated with the physical stability of the site, the NRC staff have concluded that the 500-year lifespan of the surface cover is important but not essential for this safety objective. Physical stability is provided by the tanks in combination with the infill grout; however, NRC staff concluded in Section 3.12.2 in this report that DOE should complete a structural stability assessment. The surface cover would provide some additional protection during a driller exposure scenario, but the thickness of engineered surface barrier does not significantly affect the dose to a receptor exposed to waste brought to the surface by water well drilling. The engineered surface barrier is essential to preventing waste from being exposed by human excavation. As discussed in Section 3.3.2.1 of this report, a minimum of 3 m (10 ft) of soil remains after 10,000 years even with a worst-case potential wind erosion rate of 15 cm (6 in) in 500 years such that intrusion into the waste by excavation can be excluded.

Although the engineered surface cover, based on the current preliminary designs presented in the WMA C PA (2016), can store water for evapotranspiration and decrease infiltration and thereby recharge for its 500-yr lifespan, the NRC staff concludes that it is not a risk-significant feature for meeting the performance objective §61.41 based on the results of the sensitivity analyses. The engineered surface cover is essential for maintaining a depth to waste to provide protection to an inadvertent intruder from excavation to meet the performance objective §61.42. Due to the surface cover's importance, NRC staff recommends that for the final Waste Evaluation for WMA C, DOE perform an analysis 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 (Recommendation #17).

3.4.2.1.2 NRC Evaluation of Infiltration

Water infiltration beyond the root zone, interacting with waste and carrying contaminants to an aquifer as groundwater recharge, is frequently a risk-significant process. However, multiple barriers may mask the potential effectiveness of a barrier that reduces the infiltration rates. The concrete shell and infill grout will limit the sensitivity of the surface cover to the overall performance. Consequently, based on the results of the sensitivity and uncertainty analyses, the parameters associated with infiltration through the engineered surface cover do not rank as risk-significant factors in the PA due to the modeled performance of the concrete and grout. Degradation of the concrete and grout is assumed to occur thousands of years after the engineered surface cover is assumed to fully degrade. However, if the concrete and grout were to degrade or not function as anticipated, the surface cover could be an important barrier to the release of radionuclides and the timing of peak dose.

DOE's original GRT4 sensitivity case was designed to examine the impact of degraded infill grout. With the degraded cementitious material, the original GRT4 case showed a 144% change to the maximum concentration of radionuclides at the downgradient POC, indicating that the flow and transport properties of the infill grout and concrete shell were not risk significant. DOE modified the sensitivity case GRT4 in response to the NRC RAI. As previously discussed in Section 3.4.1.2.1 of this report, the modified sensitivity case for GRT4 increased the maximum concentration at the downgradient POC for the base case, demonstrating that infiltration rates can influence the contaminant concentration and the dose. The dose increased from 1 x 10^{-3} mSv/yr (0.1 mrem/yr) to 0.022 mSv/yr (2.2 mrem/yr) or a 2,100% increase.

The Hanford Prototype Barrier (HPB) was noted by DOE as providing information that will be used to develop and implement a closure cover. The HPB has an asphalt layer (PNNL-17176, 2007). The asphalt layer may have influenced the low infiltration rates observed to date, and if it is to be included in the eventual surface barrier design degradation of bitumen from biological and other processes should be evaluated. In addition, the presence of a continuous impermeable layer can influence the observed evapotranspiration by plants as deeper-rooted plants send roots down to the impermeable layer that then branch out laterally. If the impermeable layer is degraded, the rate of evapotranspiration can decrease.

In addition, the final design for the surface barrier as well as the side-slope design and layout are unknown (e.g., if its initial surface will be vegetated or covered by gravel) so that future infiltration rates through the side slope are uncertain at this point. Table 6.1 in PNNL-16688 (2007) shows a range for gravel side slopes between 1.9 mm/yr (0.07 in/yr) for a surface with shrubs and 33 mm/yr (1.3 in/yr) for a side slope that is not vegetated. Side-slopes were observed to have over 1,000 mm/yr (39 in/yr) in some instances (Gee, 1992). Therefore, from an infiltration standpoint, NRC staff recommends that for future WIR evaluations and assessments, the 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 a technical bases for infiltration rates through side slopes should be provided (Recommendation #18).

In the absence of details about specific design information, including the shape, size, and specific components of the cover, the risk significance cannot be precisely evaluated. In addition, a quality assurance/quality control program must hinder defects and construction errors from occurring. The NRC staff concludes that DOE's approach to assessing the range of infiltration rates through the engineered surface cover is reasonable for planning purposes. The safety function of EB1, "RCRA Cover (infiltration reduction)," although not significant to total system performance due to other redundant barriers, can reduce the infiltration rates so as to limit the amount of water available to contact and transport radioactive waste during the first 500 years after closure, based on the current preliminary cover designs presented in the WMA C PA.

Summary of Review

- DOE's approach to assessing the range of infiltration rates through the engineered surface cover is reasonable for planning purposes. Although the engineered surface cover, based on the preliminary designs presented in the WMA C PA (2016), is not a risk-significant feature for meeting the performance objective §61.41 based on the results of the sensitivity analyses, but it does provide defense-in-depth. For meeting the performance objective §61.42, the engineered cover is important to maintain a depth to waste to provide protection to an inadvertent intruder from excavation. The engineered cover also provides physical stability.

- The safety function of EB1, "RCRA Cover (infiltration reduction)," although not significant to total system performance due to other redundant barriers, can reduce the infiltration rates so as to limit the amount of water available to contact and transport radioactive waste during the first 500 years after closure based on the preliminary cover designs presented in the WMA C PA.

- Recommendations #16 through #18 are discussed this section and included in Table 5-1 of this report.

3.4.2.2 NRC Evaluation of Engineered Subsurface Barriers

Engineered subsurface barriers can play an important role in limiting releases of residual waste into the environment and prohibiting inadvertent contact with buried waste. The sections that follow document the NRC's review of DOE's use of engineered subsurface barriers at WMA C.

3.4.2.2.1 NRC Evaluation of Degradation of Pre-Closure Cementitious Material and Steel

Based on the information presented and reviewed in the WMA C PA and supporting documents, and taking into consideration the multiple barriers or safety functions, NRC staff determined that

DOE has provided adequate information on the processes involved with the long-term degradation of cementitious material at the WMA C. However, NRC staff also concluded that the evaluation of unprotected carbon steel corrosion was incomplete. Further, NRC staff identified additional areas where relevant analyses or information could reduce uncertainty and provide additional confidence in the PA results.

The results of the DOE investigations into the core samples taken from tanks A-106 and SX-115 showed varying results with samples from Tank A-106 in relatively better condition than those taken from SX-115 in which cracks and air voids were found. DOE discussed possible causes in the WMA C PA and RPP-RPT-46879 (2016) for the degradation found in Tank SX-115 including variations in the quality between concrete batches, poor concrete placement during construction, and that the SX-115 tank wall concrete was subjected to higher temperatures during tank farm operations than other tank farms. The NRC staff recognizes that sampling of concrete from the buried tanks is a considerable engineering challenge and acknowledges the effort DOE put forth. As a result of the relatively limited sampling of concrete associated with the single-shelled tanks, the condition of the concrete in WMA C is uncertain. The discussion of the differences between tanks A-106 and SX-115 and the possible causes for these differences is informative. Production of concrete batches that varied in quality and poor concrete placement during construction are likely to have occurred considering the scale of the Hanford Site project and how quickly the large structures were built. Relevant available records dating from the time period are lacking and due to this relatively large uncertainty, it cannot be excluded that specific structures received inferior concrete or rebar or experienced poor concrete placement. In contrast, more information is known about which tanks were subjected to higher temperatures during tank farm operation. RPP-10435 (2002) stated concrete exposed to high temperatures could experience strength reduction. Figure 3-7 shows an example of construction defects where rebar was exposed at the time of construction. The defects were likely patched but patches tend not to adhere as well.

As discussed above, Table 6-2 in the WMA C PA included bounding depths for a carbonation front if the concrete is intact. However, rebar or protruding iron was not included in the degradation discussions DOE provided in the WMA C PA, although figures in the document did show rebar. Construction photos and engineering diagrams show dense areas of rebar throughout the designs. Corrosion of rebar or any protruding iron feature (e.g., piping, hangers) and the resulting corrosion products would occupy a greater volume than the steel, potentially causing cracking, delamination, and spalling of the concrete to occur. Because steel reinforcements typically are placed within a few inches of the concrete surface, not at half the concrete thickness, rebar corrosion-induced concrete degradation will initiate before the complete degradation of the concrete. Based on Figure 6.7 (see Figure 3-7 below) and Figure 6.9 in the WMA C PA, depths to rebar in the 100-series tank walls are approximately 63 mm (2.5 in) from the backfill and vary in thicknesses for the basemat from approximately 107 mm (4.2 in) at the center to less than a 38 mm (1.5 in) around the outer edge of the basemat.

The thicknesses between soil and rebar are similar for the 200-series tanks, thicknesses being slightly less for the walls and slightly more for the basemat. This assumes all rebar was placed perfectly according to design specifications, which is unlikely. Based on Table 6-2 in the WMA C PA, the simulated carbonation front reached a depth of 84.5 mm (3.3 in) 5,000 years after closure. This would be sufficient to reach the rebar of the tank walls and some of the rebar in the basemats.

Source: RPP-46305 | PNNL-19403, 2010, *Single-Shell Tank Inspection Program*, Rev. 0, Washington River Protection Solutions, LLC, Richland, Washington.

Figure 3-7 Example of Defects in Tank

The carbonation of the concrete would allow lowering of the pH near the rebar such that the protective magnetite layer would no longer be stable and possibly accelerate the corrosion rate of rebar. Even bounding carbonation rates less than 10 mm (0.4 in) per 70 yr would be sufficient to reach the rebar. Assuming a carbonation rate of 6.5 mm (0.26 in) every 59 yr (year of the Tank A-106 coring samples, or 2014 minus the age of the tank: 1955), would give a value of 0.85 for the constant A. After 5,000 years, the carbonation front would have moved roughly 60 mm (2.4 in) into the concrete.

Both the 100-series and the 200-series tanks include steel protrusions from the tank walls into the surrounding soils (see Figure 3-8) that are not surrounded by a protective layer of concrete. Any corrosion of these protrusions outside of the concrete walls would continue to corrode along the rebar and into the concrete walls itself. DOE has stated that corrosion of the protruding steel would be evaluated further during closure (NRC, 2018d).

Although chloride ions are not present in high amounts in the surrounding soil at Hanford (RPP-RPT-46879, 2016), DOE should include a technical discussion of the probability of rebar corrosion and the consequences of such a corrosion hastening the degradation of the concrete walls and the basemats themselves in future tank farm PAs. This would include reinforcement corrosion of exposed steel and technical basis on why carbonation-induced corrosion of reinforcement steel is not a dominant degradation mechanism for the concrete walls in the tanks.

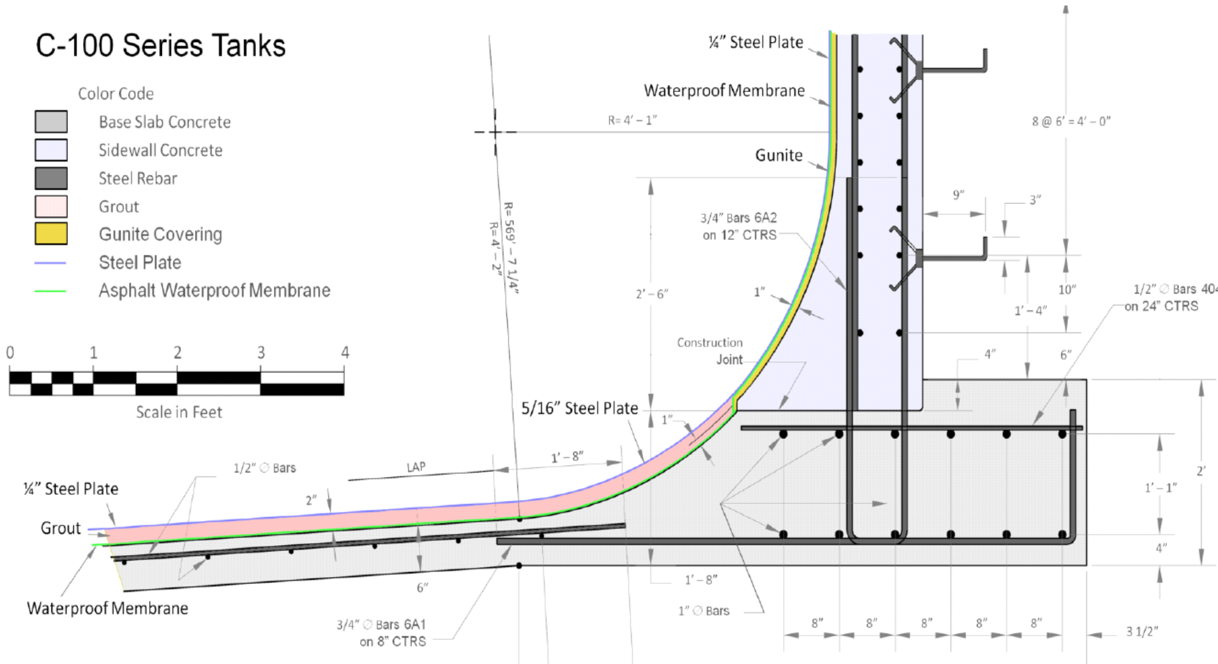


Figure 3-8 Engineered Structure at the Base of the C-100-Series Tank
[Figure 6-7 in the WMA C PA (2016)]

The NRC staff concludes that the DOE analysis of tank shell deterioration is complete for WMA C because of the risk-significance of the vault walls to overall performance, but that the evaluation is technically incomplete and results in base case results that are based on diffusion-only release and optimistic. Therefore, NRC staff recommends that for future WIR evaluations and assessments, degradation of exposed steel and rebar that impacts the properties of walls and basemats should be evaluated (Recommendation #19).

Although the basemats are conceptualized to be the main transport pathway of radionuclide release, the WMA C PA did not discuss these features of the tanks in much detail. DOE has modeled the basemats assuming no degradation, and diffusive transport is expected. Advective transport through cracks or preferential pathways through the basemat is not assumed to take place until after 20,000 years. DOE has stated that detailed information about the basemats is difficult to obtain (NRC, 2018d), and little information is known about the current state of the basemats in WMA C or at any of the tank farms, and yet considerable performance credit is attributed to this layer that was intended primarily for structural support during construction.

In RPP-ENV-39658 (2010), DOE examined leaks that had occurred at the Hanford SX-Tank Farm and the results of separate grout testing. This testing demonstrated that if grout is heated to 116 °C (240 °F), the grout beneath the steel liners in the tanks of the SX-Tank Farm would evolve water vapor accompanied by high pressures that had the potential to damage the tanks. The waste in Tank SX-113 was reported to have exceeded 116 °C (240 °F) in 1958, indicating the tank temperature was sufficient to cause vaporization of the water in the grout beneath the steel liner. The pressure beneath the steel liner could possibly be relieved through: 1) cracking of the concrete tank shell without apparent damage to the steel liner, or 2) through damage of the steel liner and venting of the trapped gases into the tank (RPP-ENV-39658, 2010).

In the WMA C PA, DOE presented the results of material properties tests of concrete cores from the haunch and wall of Tank SX-115. Although all samples were above the intended design value of 20,680 kPa (3,000 psi), the compression strength and tensile strength tests indicated a pattern of possible decreasing strength with depth. DOE states in report RPP-10435 (2002) that the most significant structural uncertainty is the condition of the reinforced concrete basemat and footing, due to the inaccessibility for inspection. Fill and empty cycles and temperature changes due to thermal and operational loads may have the potential to weaken the basemats. Given that the less-than-a-foot thick basemat is the thinnest physical barrier between the waste and the soils and given the lack of information on the current state of the basemats, DOE should include a technical discussion of the probability of cracks and advective flow through the basemats in future evaluations and should also include a technical basis supporting their assumption of 10,000-year degradation-free basemats. Therefore, NRC staff recommends that for future WIR evaluations and assessments, basemats should be assumed to be free of cracks only if characterization data is available (Recommendation #20).

DOE discussed alkali-aggregate reactions in the screening assessment that was completed for potential degradation mechanisms of concrete. Though alkali-aggregate reactions have not been observed in the limited systems that have been evaluated by DOE, if it does occur alkali-aggregate reactions have the potential, when a threshold is reached, to severely degrade materials in a short amount of time relative to the timeframes considered in a PA. The phenomenon was not well-known when the Hanford Site was constructed and so would not have been a design consideration. Without additional sampling and testing of WMA C structures, DOE should include the uncertainty associated with buried concrete performance in the base case PA. Additional research may allow for the elimination of this FEP from inclusion in the base case.

In the base case, DOE assumed the effective diffusion coefficient of the basemat was $3x10^{-8}$ cm^2/s. The basemats at WMA C have not been characterized to determine their properties. Properties for basemat diffusion were based on laboratory measurements of small-scale samples (PNNL-23841). The value assigned for the basemat diffusion coefficient was the median of the laboratory-scale test results. The samples used in the experiments contained cement, fly ash, and steel fiber. The nominal water to cement ratio was 0.4. This formulation is unlikely to be representative of the materials used for the basemat concrete. In addition, properties of cementitious materials can be strongly influenced by quality assurance. WMA C was constructed in 1944 to 1945 when the quality assurance practices were generally not as robust as they are today. The basemat concrete layers were not intended to have a hydraulic barrier function (they were to provide mechanical support and a level surface for the tanks). For comparison, non-conformance reports were issued during construction of the AP tanks in the 1980's which noted hardened concrete (cold-joint formation) between layers, inadequate vibration, air voids, surface aggregates, soft patches, excessive cracking, and freezing of patches (RPP-RPT-55983). Figure 3-9 is the distribution of effective diffusion coefficients from numerous literature reports (CNWRA 2009-001). Absent measurements of actual basemat concrete at Hanford and considering the scale of the basemats and the quality assurance procedures of the 1940's, the effective diffusion coefficient assigned in future PAs should be towards the upper end of the distribution shown in Figure 3-9. Figure 3-10 provides the fluxes of ^{129}I from Tank C-110 for DOE's base case (in blue), and then an alternate case with an increase of moisture content near the tank (in orange) where the waste is located (DOE used a moisture content from the center of the tank shielded from moisture flow). The second alternative case (in gray) has increased moisture content as well as a $3x10^{-6}$ cm^2/s ($4x10^{-7}$ in^2/s) diffusion coefficient for the basemat concrete. The flux has increased a factor of 400. Therefore, the NRC staff recommends that for future WIR evaluations and assessments, DOE should use basemat effective diffusion coefficients that are higher to reflect the uncertainty about construction quality. Sampling and testing of basemat concrete or appropriately analogous materials should be considered (Recommendation #21).

DOE and the NRC staffs have previously discussed some of the technical issues documented in this section (i.e., poor quality or placement of concrete, rebar corrosion byproducts accelerating concrete degradation, and risk significance of the basemat) and although DOE may have agreed that additional information would be helpful to understanding various corrosion or degradation processes in more detail, DOE generally considered these issues not significant due to the uncertainty/importance analysis and sensitivity analysis that were carried out and subsequently documented in the WMA C PA. However, in their review, NRC staff did not identify an analysis that demonstrated the individual performance of the safety functions although the various components of the tank are treated as separate barriers and designated as safety functions. In Section H.3 of the WMA C PA, these include EB8, "Tank structure (permeability)"; EB9, "Grout in tank (permeability)"; and EB13, "Tank Base Mat (permeability)." So that the performance of the individual barriers or safety functions can be demonstrated, NRC staff recommends that for future WIR evaluations and assessments, DOE should test identified safety functions separately. Specifically, as discussed in detail below, the uncertainty associated with pre-closure cementitious material and rebar is large so that uncertainty or barrier importance analysis and sensitivity analysis should be expanded in order to analyze relevant safety functions separately (Recommendation #22).

In DOE's simulations, assumptions about tank degradation involved degradation of the entire tank structure and infill materials, which would include the concrete dome, the concrete sidewalls, floor, basemat, and infill grout performing as one barrier. A difference in the degradation rate between the grout and the concrete that allows a slower rate of grout degradation than concrete may produce higher releases. Degraded concrete is likely to have variable particle sizes, and therefore, much different moisture characteristic curves than intact concrete. Depending on the condition of the steel liner, water flowing through the concrete in the tank sidewalls could contact the residual waste on the stiffener ring or walls and at the periphery of the tank. Water in the basemat may be able to transport residual waste that has not been solidified after the grout pour. If a tank steel liner were to remain intact, it would be expected that the risk significance of such an alternative conceptual model would be minimal for that tank.

A conceptual model whereby grout degrades at a slower rate than concrete with rebar is not unreasonable. The Saltstone Disposal Facility PA is a performance-based, risk-informed analysis of the fate and transport of waste following final closure of Saltstone Disposal Facility (SDF) at the Savannah River Site in the State of South Carolina. Although there are many differences between the SDF and the WMA C (e.g., climate, wasteform, disposal structure including the use of a very low hydraulic conductivity geomembrane) there are also some key similarities.

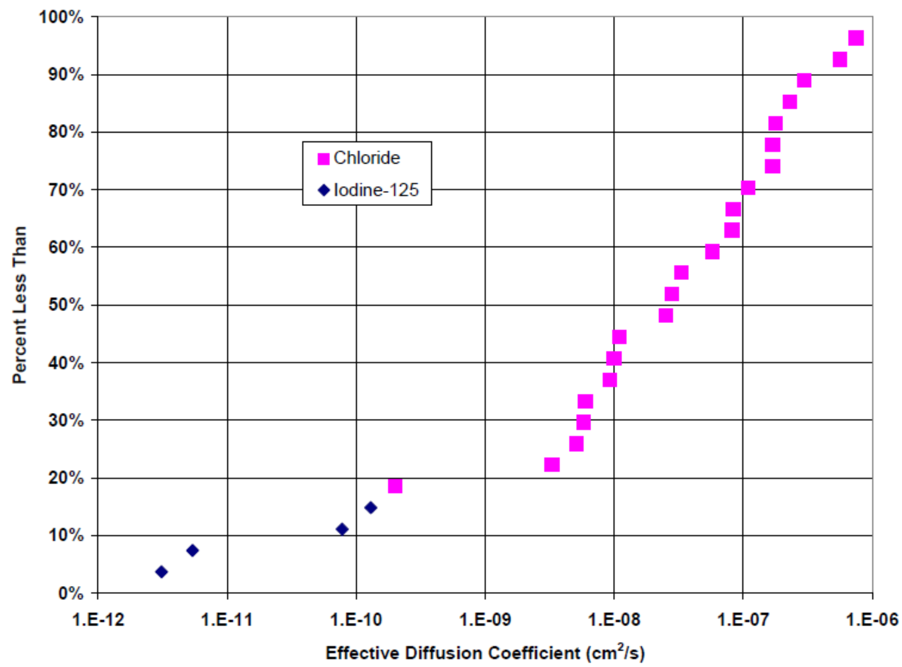


Figure 3-9 Distribution of Effective Diffusion Coefficients from Literature Sources

These similarities include concrete sidewalls with steel reinforcements and cementitious mortar on the inside with limited knowledge of the rates of change of the properties of both materials. The PA for the SDF differentiated the degradation rate between grout and concrete. As can be seen in Table 5-5 of the "Degradation of Cementitious Materials Associated with Saltstone Disposal Units" (SRNL-STI-2013-00118), the degradation of the vault roofs, walls, and floors start earlier than the Saltstone in the middle of the disposal unit. Complete degradation of the vault components occurs between 900 to 1,500 years after closure while the mortar-like Saltstone begins to degrade close to 1,000 years after closure.

Based on the degradation rates shown in Table 3-4, modeling results at 1,000 years were presented in Figure DSP-6.6 in SRR-CWDA-2014-00099 (2105). Figure 3-11 reproduces the lower right corner of the disposal unit from that figure and presents the volumetric flow rate for the selected grid zone. The vault wall is a preferred flow pathway in comparison to the cementitious mortar (i.e., Saltstone) and the lower mud mat (i.e., lower floor layer) is carrying water towards the middle of the vault.

There are differences between to the two sites, the disposal facilities, and the disposal of the waste. For example, the saturated hydraulic conductivity value of the undegraded Saltstone (6.4×10^{-9} cm/sec [2.5×10^{-9} in/sec]) allows some water to flow through the wasteform while no water is assumed to flow through the intact infill grout at WMA C (Page 6-28 in DOE, 2016). In addition, the gravel-containing backfill at WMA C has a higher hydraulic conductivity than the soils surrounding the Saltstone disposal units.

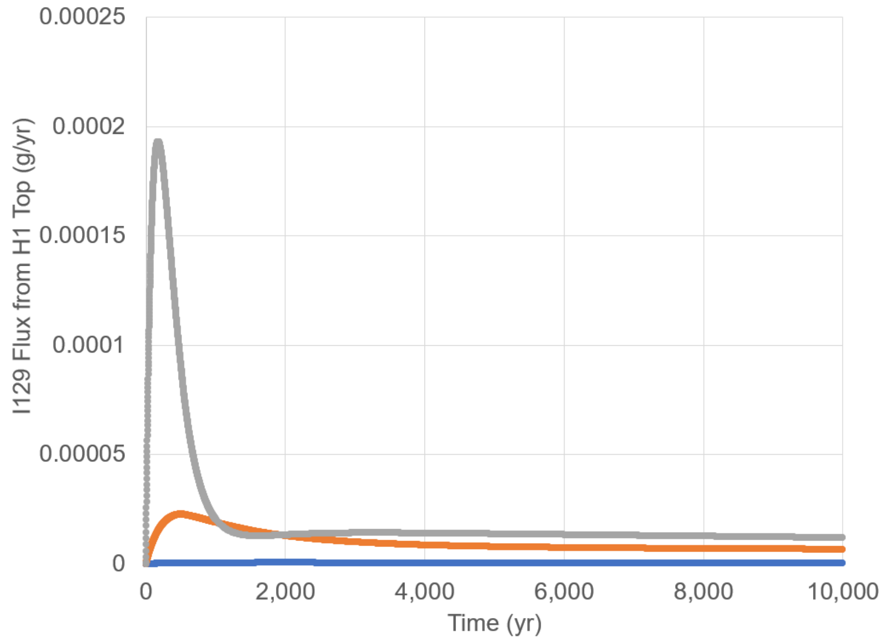


Figure 3-10 Flux of I-129 from Tank C-110 for DOE's Base Case and Two Alternative Cases

However, this alternative conceptual model, whereby infiltrating rainwater flows along the top and then along the sides of the intact grout 500 years after closure may be a plausible risk-significant conceptual model for the WMA C or future PAs of other WMAs. DOE's EHM modeling approach with uniform properties assigned to large geologic units may not be capable of capturing actual water flow around discrete units surrounded by materials with heterogenous properties. DOE identified and documented an extensive list of relevant FEPs and identified FEPs that may degrade or modify the performance of a safety function in some way. However, DOE's current safety function methodology did not appear to be able to identify interdependencies and interrelationships between FEPs that could result in plausible alternative conceptual models, such as the one discussed above.

3.4.2.2.2 NRC Evaluation of Degradation of the Infill Grout and Wasteform

As previously discussed, DOE did not include a wasteform in the PA. The radioactivity of the waste was assumed to be uniformly distributed over the plan-view area of the system being evaluated. Empirical data was used to develop release rate expressions for ^{99}Tc and uranium isotopes. NRC's review of release rates is found in Section 3.5. DOE's approach of not representing the residual waste as a wasteform in the model is acceptable because it results in a conservative estimate of release rates. The source cell in the model has no residual waste mass and, therefore, no degradation of a wasteform. Degradation of the infill grout was not included in DOE's base case as no processes were deemed credible to substantially alter the material over the evaluation period. DOE evaluated some of the types of processes that NRC would anticipate could potentially alter the performance of infill grout. The grout in the PA model acts to limit advection of infiltration to the waste layer. The grout formulation has not been selected at the time of this review, therefore, NRC staff cannot reach a conclusion on the degradation evaluation of the grout.

Table 3-4 Degradation Analysis Summary for 46-meter (150-foot) Diameter Saltstone Disposal Units [Table 5-5 in SRNL-STI-2013-00118 (2013)]

SDU Feature (thickness)	Mechanism(s) for Degradation					
	Best Estimate		Nominal Value		Conservative Estimate	
Roof (8 inches)	**Sulfate**	**Carbonation**	**Sulfate**	**Carbonation**	**Sulfate**	**Carbonation**
Degradation (inches)	7.8	0.2	8	0	8	0
Degradation (years)	1,820	420	961	0	757	0
Delay (years)	0	1,400	0	1,400	0	1,400
Elapsed time (years)	1,820		961		757	
Wall (8 inches)	**Sulfate**	**Carbonation**	**Sulfate**	**Carbonation**	**Sulfate**	**Carbonation**
Degradation (inches)	7.7	0.3	7.7	0.3	8	0
Degradation (years)	1,797	897	922	22	757	0
Delay (years)	0	900	0	900	0	900
Elapsed time (years)	1,797		922		757	
Floor/upper mud mat (12 inches)	**Sulfate**	**Carbonation**	**Sulfate**	**Carbonation**	**Sulfate**	**Carbonation**
Degradation (inches)	11.6	0.4	11.8	0.2	12	0
Degradation (years)	2,717	1,317	1,413	13	1,135	0
Delay (years)	0	1,400	0	1,400	0	1,400
Elapsed time (years)	2,717		1,413		1,135	
Saltstone (22 feet)	**Decalcification**		**Decalcification**		**Decalcification**	
Degradation (feet)	22		22		22	
Degradation (years)	2,240,856		224,086		22,409	
Delay (years)	1,820		961		757	
Elapsed time (years)	2,242,676		225,047		23,165	
Column (2 feet, length)	**Carbonation**		**Carbonation**		**Carbonation**	
Degradation (feet)	2		2		2	
Degradation (years)	29,283		584		234	
Delay (years)	1,820		961		757	
Elapsed time (years)	31,103		1,545		991	

The grout formulation has not been selected at the time of this review, therefore, NRC staff cannot reach a conclusion on the degradation evaluation of the grout. Some processes that were not explicitly included in the base case assessment, but could be important, should be evaluated in research completed to establish the tank infill grout formulation.

The potential presence of organic compounds in the waste was raised in NRC's RAI and addressed with a sensitivity case in DOE's response to the RAI. Limited research has been completed to evaluate the impact of organic substances (e.g., kerosene, tributylphosphate (TBP), di-(2-ethylhexyl) phosphoric acid (D2EHPA), ethylene diamine tetraacetic acid (EDTA), hydroxyethylene diamine triacetic acid (HEDTA)) on the performance of cements.

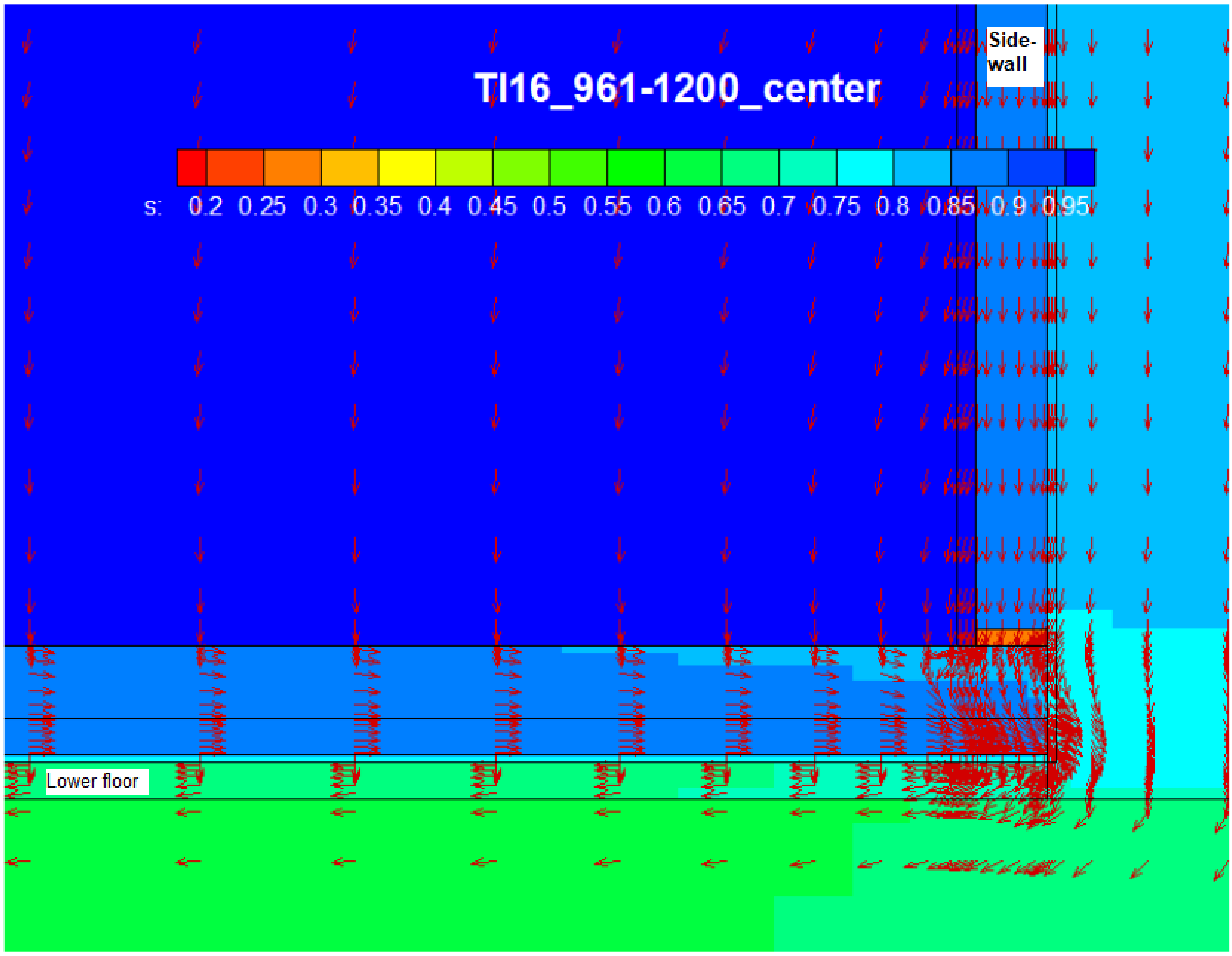


Figure 3-11 Flow Rate Transient and Flow Field Views at 1,000 Years for the 46-meter (150-foot) Diameter Saltstone Disposal Units
[Figure DSP-6.6 in SRR-CWDA-2014-00099 (2015)]

Cements made with TBP and its degradation products showed decreased performance with high heat, and acid attack lead to swelling which could result in cracking in a confined situation (El-Dessouky et al., 2001). The presence of EDTA was found to retard the curing/hydration process (Thomas, 1983), and EDTA was recovered from cement pore fluids after 6 months of curing (Smillie, 1999). Studies with HEDTA found it could shorten time to cracking and reduce compressive strength (Tonini, 1978). DOE attempted to add Portland cement to Tank BY-105 in the 1970's, and it was reported that the cement did not set due to the high-caustic, high-salt environment (Agnew, 1997). The grout formulation selected will need to be tested to ensure it is compatible with the tank residuals at Hanford.

DOE mentioned the FEP shrinkage in Section 6.2.1.2.1 of the WMA C PA. However, it was not carried forward in the evaluation, nor was a basis provided for its elimination. DOE has been working on developing grout formulations for the closure of tanks (WRPS Closure Presentation 2/21/2019 – (Hendrickson, 2019)). The amount of shrinkage for formulations evaluated in those studies was 1.4 to 6.0%. The 100-series tanks have a diameter of 23 m [75 ft], such that at the low end of 1.4%, the shrinkage gap would be almost 16 cm [4 in] if left unaddressed. This would allow for a potentially significant advective pathway for release, considering the inlet ports in the tanks are not sealed and a significant amount of water leaks through the inlet ports on some tanks in the present day (RPP-RPT-29191). Most of the residual waste in the tanks is at the outer edge of the bottom of the tank where this shrinkage gap could occur. In addition, some tanks will have various structures and equipment left in place. If the grout does not form a tight bond with these structures and equipment after curing, shrinkage could allow potential advective pathways to occur. Figure 3-12 shows photographs of some of these features (RPP-RPT-42323, Rev.3, 2015). DOE should demonstrate that shrinkage can be adequately addressed through design and implementation. Experiments of proper scale may be needed to validate the infill grout formulation when it is completed.

Summary of Review

- Due to the overall safety margins in the results analyzed, NRC staff determined that DOE has provided adequate information on the processes involved with the long-term degradation of cementitious material at the WMA C. A combined sensitivity case presented in DOE (2019) included increasing the hydraulic conductivity to match that of backfill for all the tank components, including the pre-closure concrete with rebar, so that advective flow is present in and around the tanks at the time of closure. Results indicated that the performance objective would be met.

- The NRC staff concludes that the degradation of pre-closure cementitious material and rebar steel at the C-Tank Farm as discussed in the WMA C PA (2016) is a potential risk-significant factor for performance. Uncertainties include the concrete quality produced during the construction of the C-Tank Farm, the placement of concrete during that construction, potential steel corrosion byproducts accelerating concrete degradation, and the lack of data with regards to the current conditions of basemats.

- The NRC staff concluded that the risk-significance of each separate safety function (i.e., EB8, “Tank structure (permeability)”; EB9, “Grout in tank (permeability)”; and EB13, “Tank Base Mat (permeability)” is not known since all features were assumed to degrade at the same rate and no sensitivity analyses were performed that tested the safety functions separately.

- Recommendations #19 through #22 are discussed this section and included in Table 5-1 of this report.

Figure 3-12 Features and Structures Remaining in Waste Tanks

3.5 Radionuclide Inventory, Source Term Release, and Near-Field Transport

Radionuclide inventory, source term release, and near-field transport represent how much waste is present and at what rate it may enter the natural environment from the engineered systems in which it is presently contained. For this report, near-field transport is defined as transport within the engineered components and on or near the periphery of engineered components that influences advection or diffusional release from those engineered components.

3.5.1 Summary of DOE Analyses of Radionuclide Inventory, Source Term Release, and Near-Field Transport

Development of the residual waste inventory (volume and concentrations) remaining in the tanks and ancillary equipment in WMA C is one of the most important steps of the waste evaluation process. Impacts to public health and the environment are generally directly proportional to the concentrations of radionuclides remaining in the WMA C systems and their release rates to the environment. The sections that follow provide DOE's approach and estimates of residual waste volumes and concentrations and the associated release rates of radioactivity to the environment. The NRC staff's review and conclusions follow the summary of DOE's information. As will be discussed, development of the inventory of key radionuclides can be quite complex when sampling results are not available. The inventory is the waste that remains after key radionuclides have been removed.

3.5.1.1 Residual Waste Inventory

The primary reference for inventory information is RPP-RPT-42323 Rev. 3 "Hanford C-Farm Tank and Ancillary Equipment Residual Waste Inventory" (RPP-RPT-42323, 2015). This report documents the methodology and information used to develop residual waste inventory estimates for tanks and ancillary equipment. The report provides the assumptions, technical basis, and uncertainty in the estimates. DOE developed values for Hanford WMA C tank and ancillary waste inventory estimates (RPP-RPT-42323, Rev 3, 2015). Figure 3-13 provides a plan view of WMA C and the potential sources of residual wastes. The main sources of residual wastes are the twelve 100-series tanks, four 200-series tanks, the 244-CR vault, the C-301 catch tank, diversion boxes, and over 11 km (7 miles) of underground piping (not shown). At the time of development of the Draft WIR Evaluation, the systems within WMA C were in a variety of different states of retrieval. These states were:

- Retrieval complete, residuals sampled
- Retrieval complete, residuals not yet sampled
- Retrieval in progress, residuals not yet sampled
- Retrieval not yet started, residuals not sampled
- Retrieval not planned, sampling not planned

Different methods were used to estimate residual waste inventory remaining in different systems depending on the state of retrieval. The estimates are based upon a combination of sampling, modeling, process knowledge (expert judgment), and assumptions. Each of the methods (and their combinations) have uncertainty associated with them. DOE estimated the uncertainty in the inventory values and evaluated the impact of those uncertainties with additional sensitivity cases using the PA model.

DOE used a system called the BBI, to provide one set of inventory estimates that could be used throughout the Hanford Site in the diverse activities that occur there at any given time. The values in the BBI are developed from the information types described above. Results of sampling are preferred but they are not always available. Modeling plays a key role in some cases. Early estimates were mostly based on process knowledge supplemented with modeling (e.g., with early versions of the Hanford Defined Waste model). The Hanford Defined Waste model (HDW), Hanford Tank Waste Operations Simulator (HTWOS), and the camera/computer-aided-design modeling system (CCMS) are used for various wastes in different states of retrieval (RPP-19822); (RPP-RPT-39908); (RPP-52784). In addition, some radionuclides are not characterized when physical samples are analyzed in the laboratory. Modeling, process knowledge, and assumptions are used to estimate the inventory of those radionuclides that are not analytically characterized.

Inventory estimates were developed prior to waste retrieval operations commencing in order to help plan the retrieval operations. However, retrieval did not target specific radionuclides but rather attempted to remove as much bulk waste as possible. The inventory of 25 chemicals and 46 radionuclides were estimated. Samples, process knowledge, model estimates, surface level measurements, and waste transfer records provided the basis for the initial estimates. After retrieval, sampling was performed, or planned to be performed but not yet completed, and in-tank video and computer-aided design estimates were used to estimate the residual waste inventory. For tanks that had not yet been sampled, the inventory estimates were based on the values in the BBI combined with model estimates generated with HTWOS. For ancillary equipment (e.g., pipelines, diversion boxes, Catch Tank C-301), the average concentrations of radionuclides for the overall WMA C in the BBI were used along with assumptions about the volume of waste that would remain in each component after retrieval. In some cases, such as for the pipelines, DOE did not plan any additional waste retrieval.

At the time that the PA was being developed, waste had been retrieved from 13 of 16 single shell tanks and was in progress for the remaining three (C-102, C-105, C-111). The primary chemical and radionuclide constituents evaluated in RPP-23403 “Single-Shell Tank Component Closure Data Quality Objectives” were provided in Tables 3-8 and 3-9 of the PA. DOE had to make assumptions to develop the inventory estimates including:

- For tanks that had been retrieved the residual inventory was based on the values provided in the retrieval reports. Retrieval reports were developed and provided to the State to provide a basis for terminating waste retrieval activities.
- Radionuclide concentrations were decayed to January 1, 2020.
- For tanks not yet retrieved, the minimum volume remaining was assumed to be the retrieval goal of 10 kL (360 ft^3) for 100-series tanks and the current BBI estimate provided the maximum volume remaining.
- Waste concentrations in ancillary equipment was assumed to be represented by the average concentration of all wastes remaining in WMA C.
- Waste in C-301 and the 244-CR vault will be retrieved, removing 90% of the currently estimated waste volumes.
- Wastes either have been flushed or will be flushed from pits and diversion boxes, leaving a 0.04 cm (0.016 in) thick layer on concrete surfaces.
- Waste transfer lines are 5% full of waste, except for a plugged line and cascade lines which are assumed to be full.

A variety of different waste types, in terms of chemical and radiological composition, were sent to the tank farm during operations. These wastes were transferred between tanks and mixed with other wastes. Table 3-5 is the overview of waste types received in the 100-series tanks at WMA C (Table 3-10 from the PA). There are 10 major waste types and a variety of other wastes that have been transferred to WMA C throughout the operational history. Mixing and transfer of different wastes can result in complex physical and chemical processes making predictions of compositions difficult without analytical sampling. DOE recognized this challenge and therefore post-retrieval inventory estimates were based on tank characterization samples combined with volume estimates for retrieved tanks with post-retrieval sampling. The base case inventory assigned in the PA was based on the average BBI estimate. The BBI estimate includes information for all different types of sources of information (e.g., sampling, model estimates, templates). Upper limit concentrations were provided in RPP-RPT-42323, Rev 3 (2015).

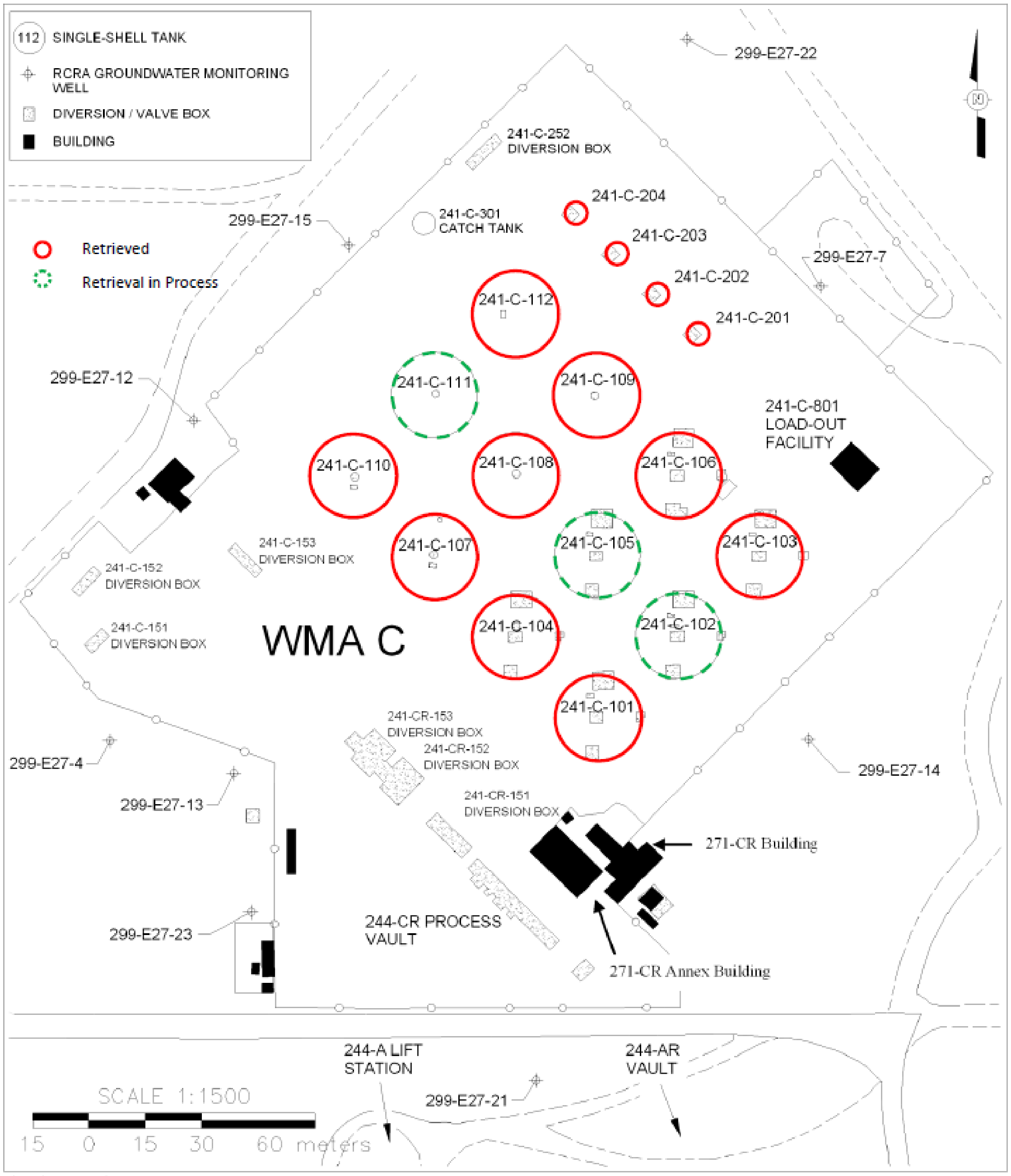


Figure 3-13 Map of WMA C Showing Locations of Potential Sources of Residual Waste
(RPP-RPT-42323, Rev. 3, 2015)

For retrieved tanks without post-retrieval sampling at the time of development of the Draft WIR Evaluation (C-101, C-107, C-112), DOE used different approaches for estimating inventory. For Tanks C-101 and C-107 the inventory estimates were based on pre-retrieval sample results, sample-based templates, and process knowledge. For Tank C-112, the inventory estimates were based on in-process transfer samples of the primary waste type in the tank and process model templates. DOE did not estimate uncertainties for these tanks. DOE indicated that they believed the inventory of soluble constituents would be lower after retrieval, and that inventory estimates would be adjusted, if necessary, after post-retrieval sampling is completed.

For tanks undergoing retrieval, the final waste volumes were not known at the time of development of the Draft WIR Evaluation. The lower bound residual waste volume was assumed to be 10.2 m^3 (360 ft^3) because initial results suggested it would be difficult to remove more waste than this. The HTWOS model was then used to estimate the composition of waste remaining by simulating retrieval operations. The upper bound volume of waste in the PA was established if no additional waste was retrieved. In the Draft WIR Evaluation, the residual inventory of waste remaining in tanks was updated based on waste retrieval and sampling performed after the completion of the PA. Waste was retrieved from six additional tanks (C-101, C-102, C-105, C-107, C-111, C-112) and samples were obtained for those tanks except C-105. The changes in tank inventory estimates were:

- The radionuclide inventory for Tank C-101 increased by approximately a factor of 3. The inventory for most radionuclide increased.
- The inventory for Tank C-102 increased significantly. The inventory of most radionuclides increased.
- The inventory for Tank C-107 was approximately a factor of 2 lower. The inventory of key radionuclides decreased.
- The inventory of Tank C-111 inventory decreased by approximately an order of magnitude. The inventory of key radionuclides that contributed to potential intruder doses decreased by various amounts (ranging from a factor of 7 to a factor of 50).
- The inventory of Tank C-112 increased by approximately a factor of 30. The quantities of ^{99}Tc and ^{137}Cs decreased whereas the inventory of ^{90}Sr, ^{241}Am, and ^{239}Pu increased.

DOE provided a table in the Draft WIR Evaluation (Table 2-5) to show the differences in pre- and post-retrieval inventories. The data indicated that the overall inventory was lower, but that the inventory of specific radionuclides in specific tanks decreased or increased. DOE also indicated that based on the changes to the radionuclides that were the biggest contributors to offsite and intruder doses, the impacts on the conclusions of the PA would be negligible. Table 3-6 provides a summary of the inventory information used in the PA.

Uncertainty in the inventory of most tanks was calculated from measured densities, radionuclide concentrations, and residual waste volumes as described in RPP-RPT 42323, Rev. 3. Radionuclide concentrations were reported on an activity per unit mass basis for most radionuclides. In waste tanks, the waste can be separated into different phases (salt solution, sludge). DOE used the restricted maximum likelihood method (REML) to estimate the mean and standard deviation of the mean for each analyte in each waste phase (RPP-6924). The overall relative standard deviation (RSD) in the inventory was then the product of the estimated

Table 3-5 Waste Types Received in 100-Series Tanks (DOE, 2016)

Year	C-101	C-102	C-103	C-104	C-105	C-106	C-107	C-108	C-109	C-110	C-111	C-112
1956	TFeCN			CW	CW					OWW	OWW	TFeCN
1957			PSN	CW	CW	PSN/ OWW		TFeCN	TFeCN		CW/ TFeCN	TFeCN
1958				CW	CW							
1959					CW				CW		CW	
1960	CW	CW	CW		CW			CW			CW	CW
1961		CW					CW	CW			CW	HS
1962		CW					CW		HS		HS	HS
1963	PSN	CW	PSN		PSN	PSN					HS	
1964	PSN	CW					HS		HS		HS	
1965		CW	PSN				HS	HS	HS			
1966		TH/CW	PSN				BNW/HS		HS			
1967		CW					HS					
1968		CW/OWW			PSN							
1969		OWW		OWW	PSN	PSS						
1970			IX	TH/OWW/PSN	PSN/RSN	PSS	IX	OWW/IX	IX	IX		IX
1971			IX	CW/OWW	PSS	PSS						
1972			CW/OWW	CW/OWW	PSS					IX		
1973			Misc	Misc	PSS		Misc	Misc				
1974			Misc	Misc	PSS	BL						
1975			Misc	Misc	PSS	BL						
1976			Misc	Misc	PSS	BL						
1977						BL						
1978						BL						

Definitions:

Colors in table are used to highlight each waste type

- **BL** B Plant strontium processing wastes and miscellaneous wastes
- **CW** Cladding (coating) waste from Plutonium Uranium Extraction (PUREX) or Reduction-Oxidation (REDOX) Plants
- **HS** 201-C Hot Semiworks waste
- **IX** Cesium denuded waste from ion exchange process in B Plant
- **Misc** Sources may include research waste from Battelle Northwest (i.e., BNW) which is now Pacific Northwest National Laboratory, reactor decontamination waste, etc.
- **OWW** Organic Wash Waste from PUREX Plant
- **PSN** PUREX high-level waste (HLW) supernate
- **PSS** PUREX Sludge Supernate derived from washing PUREX HLW sludges in 244-AR Vault or 241-A and 241-AX tanks
- **RSN** REDOX HLW Supernate
- **TFeCN** Ferrocyanide waste from 244-CR vault treatment of tributyl phosphate waste
- **TH** Thorium process waste from PUREX Plant

RSDs in each of three components: concentration, density, and volume for each waste phase. The uncertainty assigned to the inventory in the PA was then based on plus and minus 2 RSDs from the mean value.

DOE provided the inventory used in Table 2-5 of the Draft WIR Evaluation, and in Tables 4-3, 3-13a, 3-14a, and 3-15a of the WMA C PA. In the RAI, the NRC staff noted that there were differences in the values provided in the tables. In response to the RAI, DOE clarified that Table 3-13a, 3-14a, and 3-15a of the WMA C PA had the correct values and the other tables were in error. Table 3-7 provides a summary of the radionuclide inventory for select radionuclides. The staff review and conclusions is based on the inventory provided in Tables 3-13a, 3-14a, and 3-15a of the WMA C PA.

DOE indicated that sample measurements are the preferred source of waste concentration information but in some cases sample data was not available. The inventory report was developed in 2015 and there were limitations to sampling at that time. Tanks C-101 and C-102 were sampled before retrievals started and were only analyzed for a few constituents. Analysis for radioactivity was not performed on C-105 samples, and post-retrieval samples for C-112 had not been analyzed. To fill the sampling gaps, process knowledge and model-based estimates were used. Information was available from historical sampling, however the historical sampling used less rigorous quality assurance/quality control (QA/QC) and so the data was less reliable.

Post retrieval sampling utilized a data quality objective process (RPP-23403) to obtain and characterize waste residual samples. Reports were generated for each tank that was sampled documenting the results of the analytical characterization (for example, see RPP-RPT-55492, Rev 1). Sample analyses was completed following RPP-PLAN-53814. For most tanks nine grab samples were obtained which were composited into three samples. Various analytical methods were used to determine the concentrations of radionuclides. The average inventory after sampling was calculated using the Best-Basis Inventory Management (BBIM) tool (RPP-5945). This tool uses the analytical sample results, the HDW model, process knowledge, or an "average" of current values in the database to provide the inventory outputs found in the inventory estimates for component closure reports such as RPP-RPT-55803.

DOE used sample-based templates from review of sample data for tanks with similar process histories and similar waste layers. The sample templates were described in the report RPP-8847, "Best-Basis Inventory Template Compositions on Common Tank Waste Layers." The decision to include tank data in a template was based on tank transfer records for the expected waste type and depth in the tank. The HDW model was used for many of the radionuclide inventories found in the BBI (RPP-19822). The HDW model (Rev. 4) is composed of four main parts: compilation of waste transfer records up to 1994, solids histories based on waste additions, calculation of supernate blending and concentrations, and combination of process information and transaction information (Agnew, 1997). The HDW model combines tank waste transfer and process information with irradiated fuel and separation plant process records from the ORIGEN2 model. HTWOS is the process model used to estimate the effectiveness of waste retrieval information. It is used to simulate retrieval operations considering the mobility and composition of waste and retrieval fluids. Because post-retrieval sampling and volume estimation are used, HTWOS does not play a significant role in final waste inventory estimates for the Draft WIR Evaluation.

Table 3-6 Summary of WMA C Inventory

Component	Volume (kL (kgal))	Chemicals (kg)	Radionuclides (Ci)[a]
Tanks retrieved or under review[b]	178 (47)	$1.06x10^{+5}$	$2.16x10^{+5}$
Tanks not retrieved or in-process (best estimate)[c]	190 (50)	$1.18x10^{+5}$	$6.26x10^{+5}$
Catch tank C-301 and CR-244 vault	8.1 (2.1)	$5.51x10^{+3}$	$1.43x10^{+4}$
Pits	0.1 (0.03)	82	210
Diversion boxes	0.2 (0.06)	160	410
Pipelines	6.1 (1.6)	$4.12x10^{+3}$	$1.07x10^{+4}$

[a] To convert to Bq multiple by $3.7x10^{10}$
[b] Includes tanks C-101, C-103, C-104, C-106, C-107, C-108, C-109, C-110, C-112, C-201, C-202, C-203, C-204
[c] Includes tanks C-102, C-105, C-111

The CCMS is a key tool DOE developed to provide more accurate estimates of the volume of waste residuals remaining in retrieved tanks, including the uncertainty in the remaining volume. The CCMS approach is described in RPP-31159, "Post-Retrieval Waste Volume Determination for Single-Shell Tank 241-C-103". The system uses in-tank videos of recorded after waste retrieval. The videos document the location of residual solids and liquids. Computer aided design three-dimensional software is used to develop three-dimensional models of the tank interiors. Knowledge of the dimensions of various tank features are used to provide scale and obtain estimates of the area and height of waste remaining in the tanks. An example of a volume contour map for Tank C-103 is shown in Figure 3-14 (Note: most of the waste is on the periphery of the tank). The CCMS method was also used to develop a volume uncertainty equation (RPP-RPT-22891).

3.5.1.2 Residual Ancillary Equipment Waste Inventory

Ancillary equipment includes catch tank C-301, the 244-CR vault, pump pits, diversion boxes, and pipelines. DOE indicated that little information was available for the waste in the catch tank and the waste transfer pipelines. The waste volumes at the time of the analysis for the C-301 catch tank and the 244-CR vault were based on measurements. As previously indicated, DOE assumed that 90% of the waste in these components would be retrieved prior to closure. The radionuclide concentrations prescribed for the waste was based on the average measured concentration of all waste remaining in WMA C. For the waste transfer pipelines, pits, and diversion boxes both the volumes and concentrations were unknown. The residual waste in the pits and diversion boxes were expected to be small. DOE assumed that the pits and diversion boxes either have been or will be flushed leaving only a thin layer (0.04 cm [0.1 inch]) of material over the surface area. The total waste in the pits was estimated to be 100 L (30 gal). The total waste remaining in the diversion boxes was estimated to be 200 L (60 gal).

Catch tank C-301 is an underground tank characterized as a miscellaneous storage tank. The tank received wastes from the 241-C-151, 241-C-152, 241-C-153, and 241-C-252 diversion boxes. It is a cylindrical tank with 13 cm (5 in) thick reinforced concrete walls. The outside diameter of the tank is 6.4 m (20.8 ft) and the total height is 5.8 m (19 ft), though the waste can only reach a height of ~4.6 m (15 ft) before it would flow out the inlet pipes. According to RPP-RPT-45723 the tank contained a layer of sludge 1.17 m thick (3.8 ft) as of June 1985.

Table 3-7 Summary of WMA C Inventory – Select Radionuclides (Curies)[a]

System	^{99}Tc	^{129}I	^{3}H	^{137}Cs	^{90}Sr	^{237}Np	^{241}Am	^{239}Pu	^{240}Pu	^{238}U	^{233}U
C-101	4.34E-02	5.55E-05	2.45E-02	3.61E+02	3.29E+03	3.45E-04	9.91E+00	1.83E+01	1.96E+00	1.72E-01	1.71E-07
C-102	3.56E-03	2.56E-03	2.15E-05	8.07E+01	2.94E+02	5.16E-05	2.12E+01	6.49E+01	1.55E+01	9.78E-02	2.17E+00
C-103	4.48E-02	3.00E-03	3.98E-03	6.07E+02	6.78E+03	1.35E-02	4.83E+00	4.99E+00	1.04E+00	1.64E-02	5.85E-03
C-104	3.04E-01	4.84E-04	9.32E-03	6.22E+02	4.89E+03	7.97E-02	8.46E+00	5.15E+00	1.55E+00	4.39E-01	2.18E+00
C-105	7.81E+00	8.93E-03	4.08E+00	5.07E+03	2.88E+04	1.93E-04	2.83E+01	5.27E+01	1.04E+01	2.44E-01	5.01E-07
C-106	1.64E-01	6.31E-04	4.17E-03	1.00E+03	4.50E+04	5.41E-02	6.38E+01	1.67E+01	3.57E+00	9.02E-04	1.82E-03
C-107	2.14E+00	4.07E-02	1.44E-02	2.32E+03	2.42E+04	2.08E-04	3.70E+02	1.30E+02	1.42E+01	2.11E-01	2.15E-07
C-108	4.87E-02	3.81E-05	1.94E-02	8.57E+01	1.25E+03	2.17E-05	9.46E-01	6.68E-01	7.27E-02	4.03E-02	4.10E-08
C-109	8.77E-03	2.65E-05	3.51E-03	4.31E+01	2.33E+03	6.46E-04	3.71E-01	4.01E-01	4.36E-02	9.53E-03	9.69E-09
C-110	4.46E-02	2.65E-04	1.80E-03	2.02E+01	2.62E+03	1.09E-03	4.94E-02	1.17E+00	1.27E-01	2.59E-03	1.86E-09
C-111	2.19E+00	1.41E-02	2.58E+00	7.14E+03	3.05E+05	3.32E-03	8.32E+01	9.45E+01	1.85E+01	7.88E-01	4.80E-05
C-112	1.69E+00	3.57E-05	1.06E-02	7.66E+02	2.28E+02	1.54E-04	9.42E-01	5.79E+00	6.29E-01	4.32E-02	4.39E-08
C-201	2.63E-03	4.57E-07	1.57E-04	7.01E+00	1.71E+02	3.42E-03	2.46E+00	1.58E+01	3.40E+00	3.69E-02	1.14E-05
C-202	2.50E-03	7.35E-06	1.60E-04	6.18E+00	3.31E+02	2.90E-03	1.21E+00	1.43E+01	3.08E+00	3.28E-02	1.02E-05
C-203	2.32E-03	1.47E-05	1.31E-04	9.10E+00	1.56E+02	2.70E-05	3.16E-02	4.86E-01	1.05E-01	1.09E-01	3.37E-05
C-204	3.18E-03	3.57E-07	1.13E-04	4.13E+00	1.03E+02	2.16E-02	3.16E-03	9.84E-03	2.12E-03	8.13E-02	2.51E-05
C-301	3.70E-02	2.09E-04	2.13E-03	1.23E+02	3.11E+03	2.87E-02	5.63E+00	2.17E+01	4.68E+00	2.26E-01	1.21E-01
244-CR Vault	3.80E-02	2.15E-04	2.18E-03	1.26E+02	3.18E+03	2.94E-02	5.77E+00	2.22E+01	4.79E+00	2.31E-01	1.25E-01
Pipelines	1.12E-03	6.34E-06	6.44E-05	3.74E+00	9.40E+01	8.68E-04	1.70E-01	6.57E-01	1.42E-01	6.83E-03	3.68E-03
Total	2.17E-03	1.23E-05	1.25E-04	7.24E+00	1.82E+02	1.68E-03	3.30E-01	1.27E+00	2.74E-01	1.32E-02	7.13E-03

[a] To convert from Ci to Bq multiply by 3.7×10^{10}. The notation 1E3 is equivalent to 1×10^{3}

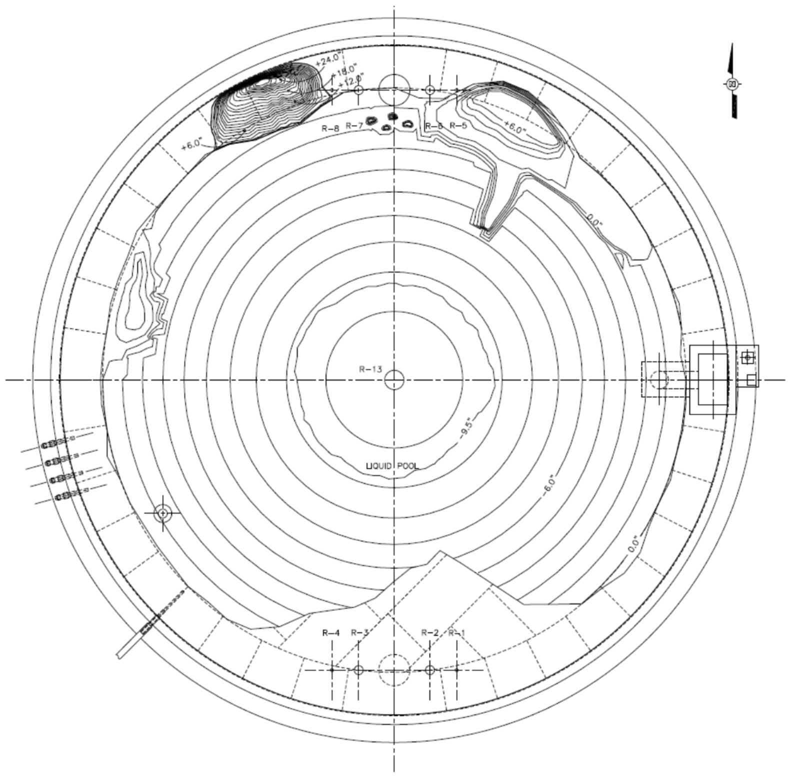


Figure 3-14 Volume Contour Map of Waste Remaining in Tank C-103 (RPP-31159)

The 244-CR vault is in the southern portion of WMA C. The vault received a variety of different wastes. The 244-CR vault is a concrete structure that is mostly underground. The vault is 31 m (102 ft) long and 7.9 m (26 ft) at the widest point. The vault contains four tanks (001 and 011 are 190,000 L (50,000 gal) and 002 and 003 are 57,000 L (15,000 gal). The waste volume estimates are based on surface level measurements and video inspections (RPP-RPT-24257). The 244-CR vault was used for scavenging Cs-137 from tributyl phosphate (TBP)-based waste. The vault was also used for uranium sludge recovery and interim storage and transfer of waste from B Plant, the PUREX Plant, and Hot Semiworks waste. DOE assumed that 90% of the waste remaining in the 244-CR vault would be retrieved and that the composition of the waste was the average of all waste remaining in WMA C.

There is approximately 11 km (7 miles) of pipelines within WMA C. There are 230 separate pipelines with different diameters and lengths (RPP-PLAN-47559). DOE indicated that after usage the pipelines were routinely flushed to remove residual waste. However, some pipelines plugged, some leaked, and characterization of pipelines in WMA C has not been completed and is not planned to be completed. DOE considered previous estimates of pipeline inventory to develop the estimate for the Draft WIR Evaluation (RPP-RPT-42323, Rev 3). These estimates ranged from 28 to 7200 L. The inventory in the WMA C PA was based on an assumed length of 11 km (6.9 miles), an assumed diameter of 10.8 cm (4.25 in), and that the pipes would be 5% full of waste. DOE indicated that they believed 5% would be conservative because of operational flushing procedures. One pipeline was known to have plugged (V122) and was assumed to be 100% full of waste. Gravity drained cascade lines that had a history of plugging and unplugging were assumed to be 100% full of waste for the purposes of the Draft WIR Evaluation. The overall result was that the residual pipeline waste volume was 6,000 L (1,600 gal). The composition of waste in the pipelines was assumed to be equal to the average of all waste in WMA C. Some pipelines are encased in other pipes (i.e., a pipe within a pipe) of different compositions. Other pipelines were in concrete encasements that were open on top; the encasement was a structural support for the pipeline. DOE did not prescribe any waste inventory to the encasements.

3.5.1.3 Source Term Release

Source term release is the representation of how radioactivity is released or made available to be transported out of the waste to the near field environment. In the case of residual waste in WMA C, the residual waste is not an engineered wasteform. Rather the waste is a complex collection of phases present from over 50 years of operations. DOE considered both mineral phase solubility-limited and matrix degradation rate-limited processes. DOE developed the conceptual models based on numerous years of testing and analysis.

As part of post-retrieval characterization, DOE evaluated the chemical and radiological characteristics of the waste and its composition, solid-phase characteristics, and leachability of contaminants of interest. Some of the reports summarizing the work included:

- PNNL-16738, “Hanford Tank 241-C-103 Residual Waste Contaminant Release Models and Supporting Data”
- PNNL-16748, “Contaminant Release Data Package for Residual Waste in Single-Shell Hanford Tanks”
- PNNL-15187, “Hanford Tank 241-C-106: Residual Waste Contaminant Release Model and Supporting Data,” Rev. 1
- PNNL-19425, “Hanford Site Tank 241-C-108 Residual Waste Contaminant Release Models and Supporting Data”
- PNNL-14903, “Hanford Tanks 241-C-203 and 241-C-204: Residual Waste Contaminant Release Model and Supporting Data,” Rev. 1
- PNNL-16229, “Hanford Tanks 241-C-202 and 241-C-203: Residual Waste Contaminant Release Model and Supporting Data”
- Deutsch, W. J., et al, 2011, “Hanford tank residual waste – Contaminant source terms and release models”

Characterization and testing of samples from these tanks formed the basis for the models developed for all tanks in WMA C. Both the main elements and the radionuclides varied significantly from tank to tank. DOE performed various analytical techniques on the samples to develop an understanding of the materials and their properties. The density of samples ranged from 1.36 to 1.77 g/cm^3 (84.9 to 111 lb/ft^3) and the moisture content ranged from 4.16% to 47.1%. Predominant metals included aluminum, sodium, iron, calcium, silicon, manganese and uranium. For tanks C-202 and C-203 the concentrations of uranium were 20.7 and 50.5 wt.% respectively, which was the highest concentration of any element in these wastes. The compositional differences between the tanks were due to: 1) mixing of various waste types, 2) chemical reactions within tanks including as a result of heat and evaporation, and 3) the effects of different waste retrieval methods. X-ray diffraction (XRD) and scanning electron microscopy/energy-dispersive spectroscopy (SEM/EDS) were used to identify phases and minerals in the residual waste. For Tank C-103, the predominant phase was gibbsite, an aluminum oxide compound.

A variety of phases were observed in Tank C-106 samples. Amorphous phases high in uranium and sodium were the predominant phases in tanks C-202 and C-203. The report (PNNL-16748) provided detailed discussions of the phases and pictures of micrographs.

Single-pass flow-through tests were used to determine the release rates of contaminants. DOE focused on the release rates of uranium isotopes and ^{99}Tc. Three different leachates were used including deionized water (DI), $CaCO_3$ saturated solution, and 0.005 M $Ca(OH)_2$ solution to represent a range of different water types that may contact the waste in the future. DOE indicated that the trends in uranium leachate concentrations for C-103, C-202, and C-203 were similar with significantly lower leaching with the $Ca(OH)_2$ solution. Grouting of the tanks is expected to result in a $Ca(OH)_2$ solution. DOE completed thermodynamic modeling to verify the results of the leach tests and the thermodynamic modeling yielded consistent results. DOE concluded that the results indicated that if the infiltrating water through the tank passes through the infill grout material, it will be conditioned to be similar to a dilute $Ca(OH)_2$ leachate solution and the uranium dissolution will remain inhibited. At some distant time in the future when the tank is sufficiently degraded such that fractures develop that do not allow appreciable residence time for infiltrating waters to contact the grout material, the leachate would be similar to the $CaCO_3$ saturated water, and at that time, the uranium concentrations may increase when the residual waste is contacted (DOE, 2016).

The source release model for uranium was developed based on the results of the experiments. DOE applied a solubility limit of $1x10^{-4}$ M for the first 1,000 years based on the assumption that amorphous uranium mineral phases control the solubility. After the first 1,000 years, a solubility limit of $1x10^{-6}$ M was applied assuming $CaUO_4$ as the solubility-controlling mineral phase under grouted tank conditions. If the grout inside a tank is degraded, then after the initial 1,000-year period a solubility limit of $2x10^{-5}$ M was applied. DOE compared the $1x10^{-4}$ M limit to the results for the material from Tank C-202 to show the model representation was conservative with respect to the experimental results (DOE, 2016).

Release rates of ^{99}Tc showed little sensitivity to the leachate type. The ^{99}Tc was released more rapidly in comparison to the uranium. The amount leached in the experiments ranged from 4.5% to 15%. To develop a model for the PA to represent the leaching of ^{99}Tc, DOE fit an exponential trend line to the fraction of ^{99}Tc remaining over time. The result was a model that had an initial release fraction that was immediately available for release of 4.5% to 15% and then a first-order reaction rate constant of $5x10^{-4}$ to $8x10^{-4}$ day^{-1}. DOE then showed how the derived model behavior compared to the experimental results (shown in Figure 3-15 (DOE, 2016)).

For all other radionuclides release modeling from the waste (wasteform) was not included in the PA model. The waste was assumed to be available for advection (if water was present) or otherwise would diffuse through the engineered materials. DOE's development of effective diffusion coefficients for infill grout and the basemats is discussed in Section 3.4.

Table 3-8 **Average Composition of Tank Samples (μg/g dry wt)**

Analyte	241-C-103	241-C-106	241-C-202	241-C-203
Al	136,000	81,699	13,600	<710
Ba	181	914	208	<142
Ca	616	46,490	14,500	3,140
Cr	193	(727)[a]	13,200	5,910
Fe	12,000	36,663	122,000	16,300
K	BDL	8,526	<15,800	<355,000
Mg	-42	3,162	2,560	-729
Mn	470	108,069	25,700	956
Na	7,840	46,720	58,800	95,800
Ni	420	5,373	9,070	510
Pb	892	4,814	7,980	5,630
Si	9,070	(4895)[a]	25,000	3,490
Sr	90.7	(493)[a]	1,510	409
^{238}U	3,730	310	207,000	505,000
^{239}Pu	8.02	27.7	435	18.2
^{237}Np	1.3	9.04	2.16	(0.0519)[a]
^{241}Am	0.053	2.05	0.449	0.014
^{99}Tc	0.231	1.14	0.149	(0.0947)[a]
^{129}I	(1.11×10^{-5})[a]	NA	NA	NA
F^-	(31)[a]	33	6,030	2,760
Cl^-	(5.4)[a]	87	161	201
NO_2	(59)[a]	<73	485	610
NO_3	(250)[a]	<70	3,540	4,840
$CO_3{}^{2-}$	BDL	39,500	12,200	49,900
$SO_4{}^{2-}$	BDL	<66	334	288
$PO_4{}^{3-}$	(66)[b]	<91	17,700	43,300
Oxalate	-	63,900	32,400	1,500

[a] 1 ug/g is equal to 0.0352 oz/ton.

[b] Value in parenthesis is the estimated quantification limit.

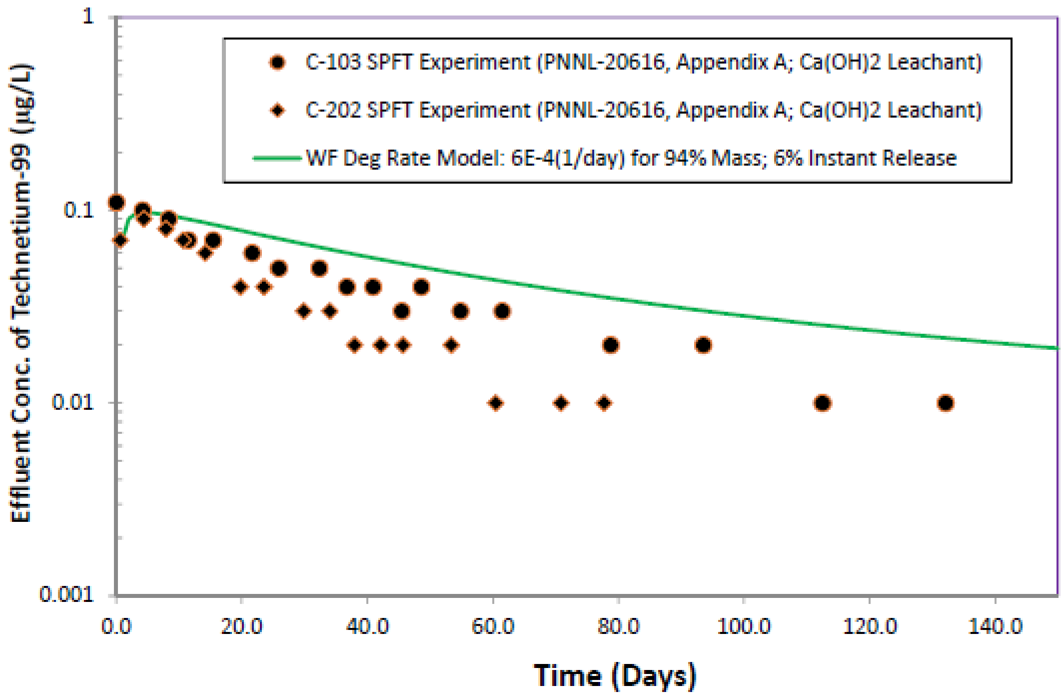


Figure 3-15 Simulated vs. Observed Results for ^{99}Tc Release

DOE's base case model for release assumes grout is present and intact that prohibits advective flow – releases are only a result of diffusion. When radionuclides are available for release, they can then partition with the engineered materials as they diffuse out of the system as a result of concentration gradients. Transport out of tanks from diffusive release was only represented in one-dimension (vertical). Lateral diffusion through the walls of the systems was not included. DOE did not include the presence of the steel tank liners in the calculations. Distribution coefficients (Kd) were prescribed for the grout and basemat as shown in Table 6-5 of the PA. Most of the Kd's were obtained from international reports (references given in the table). DOE's selection of sorption values was based on review of past reports focused developing cement sorption databases for cementitious materials, with emphasis placed on newer reports. Conditions in closed waste tanks were expected to be moderately oxidizing. When data was not available in the references DOE assigned a zero value for Kd's. If there was a large disparity between Kd's in difference references, DOE used the lower value. Near-field transport parameters were summarized in Section 6.3.1.6 of the PA.

The source-term mathematical model was implemented in GoldSim using the Contaminant Transport Module. Mass transport is modeled dynamically with compartment-based simulation. Radioactive decay and ingrowth for decay chains is considered. GoldSim provides specialized elements to model key release mechanisms including wasteform degradation, failure of barriers, and solubility controls. Both advection and diffusion are considered depending on the parameters assigned and the transport links defined. When multiple cells are connected the system of cells is simulated as a coupled system of ordinary differential equations. DOE provided the equations describing the near-field transport phenomena.

3.5.1.4 Gaseous Transport

DOE considered the potential for gases and vapors to travel upward from the residual waste through the surface barrier and to the land surface. From the land surface gaseous contaminants can enter the atmosphere and be transported to the receptor locations. Gaseous diffusion was the mechanism DOE evaluated. The partitioning of inventory into the aqueous and gaseous phases occurs within the source-term model (the waste layer). DOE considered four radionuclides that could potentially be transport in a gaseous form: ^{14}C as CO_2 gas, ^{3}H, ^{129}I, and ^{222}Rn. The Henry's law constant (K_h) is used to represent air-to-water partitioning.

The atmospheric transport pathway calculations were conducted in three steps. First, the upward diffusion flux from each source to the land surface was calculated assuming the land surface was a zero-concentration boundary. Upward diffusion from tank residuals was modeled to occur along a 10 m (33 ft) vertical pathway. Next, radionuclide transport in air was modeled using a Gaussian plume where advection and dispersion occur via wind movement to a receptor placed 100 m (330 ft) downgradient. Finally, to account for the commingling of gas plumes from different sources, DOE combined the diffusive flux from all sources into a single point source but with the release rate equal to the combine releases from all sources. The point source location was the center of WMA C which is 75 m (246 ft) from the fenceline therefore the total distance to the receptor was 175 m (574 ft). DOE applied a continual wind speed of 3.4 m/s (11.2 ft/s) and the air mixing height was assumed to be 2 m (6.6 ft). DOE described the mathematical models used and the parameters assigned. Modeling of radon transport used the same approach as for other gaseous radionuclides except radon modeling requires the selection of an emanation coefficient to represent the amount of radon that ends up in the gas phase over the total amount of radon produced. DOE selected a value of 0.2 from NCRP Report No. 103 (NRCP, 1989).

3.5.2 NRC Evaluation of Radionuclide Inventory, Source Term Release, and Near Field Transport

NRC staff reviewed DOE's development of the radionuclide inventory, source term release, and near field transport. The review covered the Draft WIR Evaluation, the PA document, many supporting references, other documents not referenced by DOE, and DOE's computational model created in GoldSim. Staff considered the guidance found in NUREG-1854 and NUREG-2175. Staff performed independent calculations with DOE's performance assessment model to develop risk insights. Documentation of the staff's review is found in the sections that follow.

3.5.2.1 NRC Evaluation of Residual Tank Waste Inventory

NRC staff reviewed many supporting reports and references to evaluate DOE's estimated residual waste inventory for WMA C. The primary reference for inventory information is RPP-RPT-42323, Rev.3. This report was well-written and clearly described the development of the inventory estimates.

DOE's approach to development of residual inventory for the tanks emphasized the use of post-retrieval sampling, when available. This is appropriate and likely to yield the results with the lowest uncertainty of the methods available. The analytical methods DOE used for sampling were appropriate. To evaluate the reasonableness of DOE's inventory estimates for the tanks, NRC staff compared the residual inventory to the total amount of inventory generated by fuel reprocessing found in RPP-13489 (2002). The fraction of waste remaining in WMA C on a radionuclide basis ranges from much less than 0.001% to a maximum of approximately 1% for a couple of the curium isotopes. The amount of ^{90}Sr was about 0.5% and the amount of ^{99}Tc was 0.04%. There were not considerable differences in terms of the percentages remaining between the inventory of isotopes generated from sampling compared to those estimated from other methods. Table 3-8 provides the percentage of the total inventory in the PA that is assigned to each component in the Draft WIR Evaluation. The Draft WIR Evaluation updated the inventory for tanks C-101, C-102, C-107, C-111, and C-112 based on retrieval and characterization information that was completed after development of the PA. The tables in the Draft WIR Evaluation did not include certain radionuclides that were included in the PA, so they were also eliminated from Table 3-9 to avoid misinterpretation of the table (^{106}Ru, ^{125}Sb, ^{134}Cs, ^{137m}Ba, ^{242}Cm, ^{90}Y). Though Tank C-106 is high in various radionuclides, the radionuclides are not the most important radionuclides from a groundwater dose pathway standpoint. Tank C-105 is high in ^{99}Tc (groundwater) and ^{3}H (air). Tank C-104 has the highest amounts of ^{232}U and ^{233}U whereas tanks C-101, C-104, and C-112 have the highest amounts of other isotopes of uranium.

Table 3-10 provides the ratios of concentrations of radionuclides to the highest concentrations observed. Table 3-9 is important for groundwater pathways exposure whereas Table 3-10 is important for intruder protection. Tank C-106 has the highest overall concentrations whereas Tank C-105 has the highest concentrations of important radionuclides. The 200-series tanks have high concentrations of some long-lived radionuclides though the volume of waste in these tanks is low. As will be discussed below, DOE developed an approach to estimate and propagate the uncertainty in the sampled inventory. However, that approach did not include some key potential sources of uncertainty.

Figure 3-16 shows the sampling locations for Tank C-109 and Tank C-110 (RPP-RPT-55492, 2013); (RPP-RPT-56796, 2014). DOE generally obtains samples from nine locations. However, because the amount of material obtained from each sampling location can be limited those nine samples are composited into three samples. The location of residual waste in the tanks is a strong function of the location of access points (inversely related). Whereas some of the sampling (e.g., C-109) was able to sample the full areal extent of the waste in the bottom of a tank, other sampling was more limited in terms of locations (e.g., C-110). Without performing a validation study by obtaining more samples from a tank and comparing the resultant inventory estimates with the inventory developed from the three samples, the representativeness of the sampling approach cannot be determined.

As described in the inventory estimates for component closure risk assessment reports, approximately 10 to 25% of the residual waste in 100-series tanks may be located on the walls and stiffener rings. This waste is not sampled. It was assumed by DOE that the waste on the walls is the same composition as the residual waste that was sampled on the floors. However due to their operational history especially with elevated temperatures, some tanks formed very dense layers of waste that were/are extremely difficult to remove. The concentrations of radionuclides in different layers can differ substantially and is modeled as differing in HDW and HTWOS. The material that remains on the floor is physically movable but cannot be removed with current pump technologies. The contaminant release data package for residual waste from tanks C-103, C-106, C-202 and C-203 describes the physical, chemical, and release properties of the waste that was sampled (PNNL-16748). The sludge remaining in tanks is highly heterogeneous in composition, structure and phase, morphology and particle size (Peterson, 2018). The sampled sludge can show considerable heterogeneity, with the density of four samples from Tank C-103 ranging from 1.36 to 1.84 g/cm^3 (PNNL-16748). How the unsampled sludge may differ from the sampled sludge is unknown. Without including this source of uncertainty, the relative importance of characterizing waste from the walls or stiffener rings cannot be established. DOE has sampled waste from tank walls at the Savannah River Site (SRS) establishing that it is technically practical to sample waste on the walls.

Analytical uncertainty from the methods used to analyze samples was not included in the uncertainty for the inventory estimates. In PNNL-16748, the authors show concentrations of radionuclides that can be significantly different depending on the method used. Results were provided for two different extraction/digestion techniques by two different laboratories. One was a fusion method and the other was EPA Method 3050B acid digestion (EPA, 1996). For Tank C-106, the radionuclide concentrations were comparable. For sludge from Tank C-202, the EPA acid digestion method resulted in ^{239}Pu concentrations that were 678% higher, ^{237}Np that was 498% higher and ^{99}Tc that was 35% lower. The reasons for the differences were not identified by the authors. The uncertainty associated with analytical methods should be included in the uncertainty in the inventory in the PA.

The NRC staff evaluated DOE’s approach to estimating uncertainty in residual waste inventory. Uncertainty in the inventory of most tanks was calculated from measured densities, radionuclide concentrations, and residual waste volumes as described in RPP-RPT 42323, Rev. 3. The REML method that DOE implemented yields RSDs that are smaller than anticipated and therefore uncertainties in residual inventory that are smaller. The source of the difference is not completely clear, outside of the fact that RSD’s calculated from the data in the sampling reports using standard equations for mean and standard deviation produces RSD’s that are 50% or more larger than produced by the REML method. Propagated over multiple parameters the impacts can be significant. Given that there are only three samples for each tank, complex methods may not be warranted. Table 2-1 of RPP-6924 shows that relative standard deviations in the bulk density of different phases was calculated by combining the means from 60 single shell tanks and then calculating the RSD in the mean of the means. It is not clear how this information is used in the estimation of the RSD for an individual tank. Pretty much every tank is different at Hanford in terms of the waste that is contained. Inter- and intra-tank variability are likely to be considerably different. Whereas one reflects the variability in operations over all the tanks, the other reflects variability in the operations for a tank. The latter quantity is what is needed for uncertainty estimates for a tank. The method DOE employed assumes that the three variables (concentration, density, and volume) are independent. Based on analysis of density and concentrations for select isotopes in sludge the three variables are not independent. For example, Figure 3-17 shows the residual volume in retrieved tanks compared to sludge density. Given that heavier particles are harder to retrieve with current pumping technology, it would be expected that there should be a positive correlation between residual tank volume and sludge density. DOE could validate the REML method by performing more extensive sampling of a tank and comparing the observed variance with the estimated uncertainty.

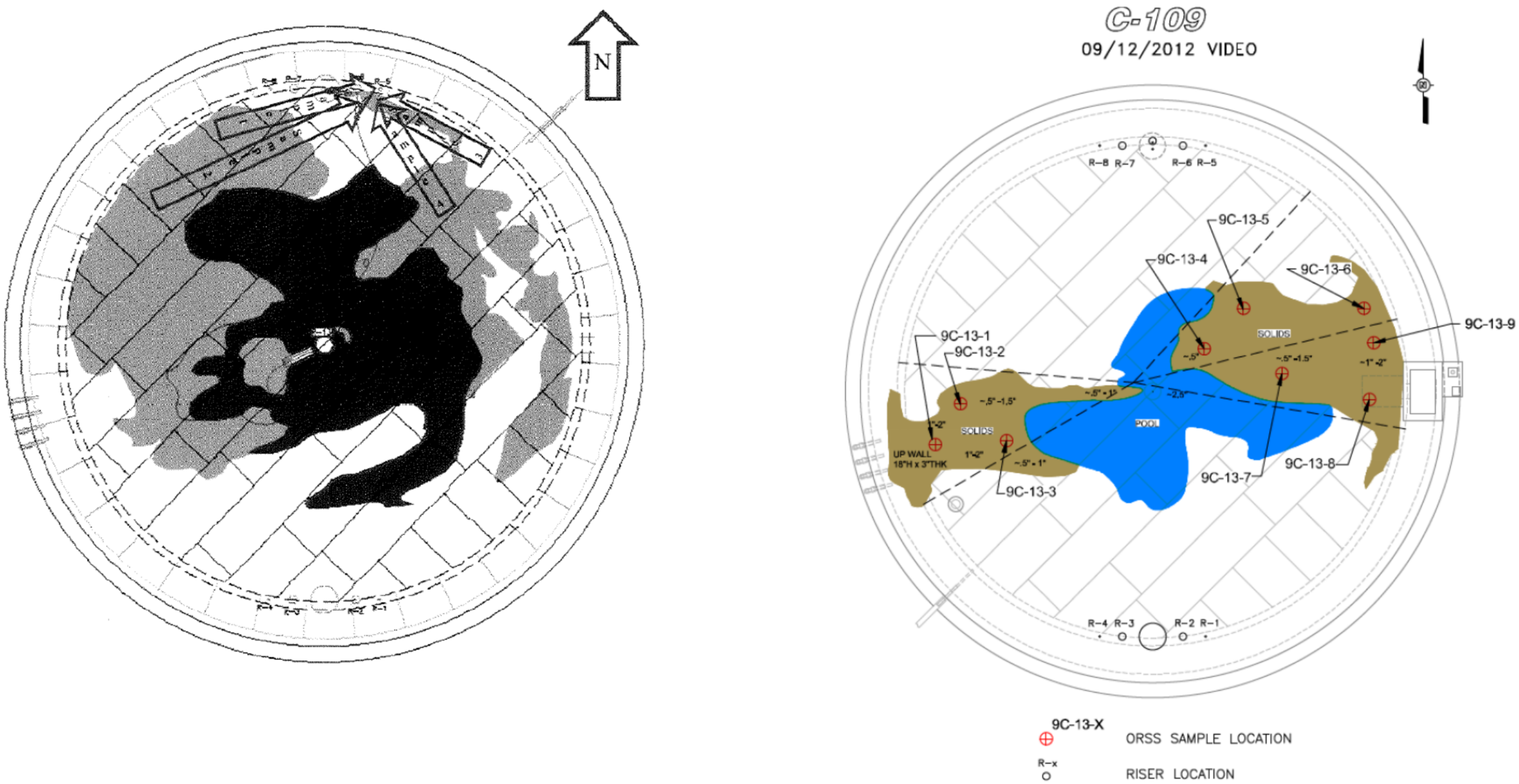


Figure 3-16 Sampling Locations for Tank C-110 (left) and Tank C-109 (right)

Table 3-9 Percentage of the Inventory in the PA Assigned to Each Component

	C-101	C-102	C-103	C-104	C-105	C-106	C-107	C-108	C-109	C-110	C-111	C-112	C-201	C-202	C-203	C-204	C-301	244-CR	DB	Pits	Pipelines
113mCd	0.1	0.2	0.6	2.0	2.3	82.3	0.1	0.1	0.1	0.0	0.0	0.1	0.0	0.0	0.0	0.0	3.3	3.4	0.1	0.2	5.1
126Sn	3.7	0.1	0.0	0.4	0.0	82.8	0.0	0.0	0.0	1.1	0.0	0.0	0.0	0.0	0.0	0.0	3.3	3.3	0.1	0.2	4.9
129I	12.6	7.4	13.9	2.2	41.5	2.9	13.8	0.2	0.1	1.2	0.1	0.3	0.0	0.0	0.1	0.0	1.0	1.0	0.0	0.1	1.5
137Cs	17.3	5.0	5.3	5.5	44.5	8.8	1.9	0.8	0.4	0.2	1.3	5.0	0.1	0.1	0.1	0.0	1.1	1.1	0.0	0.1	1.6
14C	2.3	5.5	4.6	2.0	32.2	5.5	20.9	5.4	0.5	1.0	2.6	11.6	0.5	0.1	0.1	0.1	1.4	1.4	0.0	0.1	2.1
151Sm	0.0	0.1	0.0	11.3	0.0	27.8	53.6	0.0	0.0	0.0	0.0	0.0	0.1	0.1	0.1	0.1	1.9	2.0	0.1	0.1	2.9
152Eu	0.0	0.0	0.0	1.5	0.0	85.2	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	0.1	0.1	3.6	3.7	0.1	0.2	5.4
154Eu	0.0	0.0	4.7	5.2	0.0	74.8	0.0	0.0	0.0	0.0	0.0	0.0	0.3	0.3	0.0	0.2	4.0	4.1	0.1	0.2	6.0
155Eu	0.0	0.0	4.5	2.4	0.0	78.8	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.2	0.2	0.1	3.8	3.9	0.1	0.2	5.7
226Ra	0.1	1.8	0.0	0.1	0.0	85.6	0.1	0.1	0.1	0.0	0.0	0.1	0.0	0.0	0.0	0.0	3.3	3.4	0.1	0.2	5.0
227Ac	0.1	1.1	0.0	0.5	0.0	83.7	1.3	0.0	0.2	0.0	0.3	1.0	0.0	0.0	0.0	0.0	3.2	3.3	0.1	0.2	4.9
228Ra	0.0	66.5	1.2	21.5	0.0	3.3	1.9	0.1	0.0	0.0	0.1	0.0	0.0	0.0	0.0	0.1	1.5	1.5	0.0	0.1	2.2
229Th	0.0	0.0	0.0	0.4	0.0	87.4	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	3.3	3.4	0.1	0.2	5.1
231Pa	0.0	0.0	0.0	2.4	0.0	80.3	1.8	1.0	0.7	0.2	0.4	1.4	0.0	0.0	0.0	0.0	3.3	3.4	0.1	0.2	5.0
232Th	0.0	32.9	2.4	45.1	0.0	6.8	1.0	0.2	0.0	0.0	0.0	0.0	0.0	0.1	0.0	0.2	3.1	3.2	0.1	0.2	4.7
232U	0.0	0.1	0.0	81.9	0.0	1.1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4.6	4.7	0.1	0.3	7.0
233U	0.0	10.5	0.2	73.1	0.0	0.1	0.0	0.0	0.0	0.0	0.0	1.3	0.0	0.0	0.0	0.0	4.1	4.2	0.1	0.2	6.2
234U	43.1	4.9	0.3	10.6	6.0	0.0	3.6	0.8	0.2	0.1	0.7	1.7	0.9	0.9	2.9	2.1	5.8	6.0	0.2	0.3	8.8
235U	39.9	4.3	0.4	10.2	5.2	0.0	3.2	0.9	0.2	0.1	0.5	11.3	0.8	0.7	2.5	1.8	5.0	5.1	0.2	0.3	7.5
236U	36.6	10.3	0.7	9.1	9.7	0.0	3.0	0.5	0.2	0.1	1.2	11.0	1.0	0.7	1.6	1.0	3.7	3.8	0.1	0.2	5.6
237Np	6.9	1.1	3.8	22.3	0.1	15.2	6.7	0.0	0.2	0.3	0.4	6.1	1.0	0.8	0.0	6.1	8.0	8.2	0.2	0.5	12.2
238Pu	3.0	5.4	13.0	5.9	7.5	23.7	1.0	0.0	0.2	0.2	0.8	3.6	4.4	4.0	0.1	0.0	7.5	7.7	0.2	0.4	11.4
238U	39.3	4.6	0.4	10.0	5.5	0.0	3.3	0.9	0.2	0.1	0.2	11.1	0.8	0.7	2.5	1.8	5.1	5.2	0.2	0.3	7.8
239Pu	6.4	21.1	1.7	1.7	17.7	5.6	5.4	0.2	0.1	0.4	0.8	2.1	5.3	4.8	0.2	0.0	7.3	7.4	0.2	0.4	11.0
240Pu	4.0	11.6	2.0	3.0	20.4	7.0	3.4	0.1	0.1	0.2	0.5	1.3	6.7	6.0	0.2	0.0	9.2	9.4	0.3	0.5	13.9
241Am	3.0	8.7	2.5	4.4	14.6	32.8	6.8	0.5	0.2	0.0	4.0	10.2	1.3	0.6	0.0	0.0	2.9	3.0	0.1	0.2	4.4
241Pu	5.6	14.1	1.2	7.5	11.5	12.1	0.9	0.1	0.3	0.2	0.1	6.7	5.5	4.9	0.2	0.0	8.1	8.3	0.2	0.5	12.2
242Pu	0.1	0.0	0.1	55.0	0.9	1.2	0.1	0.0	0.0	0.0	0.0	28.5	0.4	0.4	0.0	0.0	3.7	3.8	0.1	0.2	5.6
243Am	3.1	6.4	0.2	27.6	3.5	16.0	7.1	0.5	0.2	0.0	0.2	0.8	5.1	2.5	0.1	0.0	7.3	7.5	0.2	0.4	11.1
243Cm	0.0	0.0	0.0	4.4	0.0	66.5	0.0	0.0	0.0	0.0	0.0	0.0	3.7	1.8	0.0	0.0	6.5	6.6	0.2	0.4	9.8
244Cm	0.0	0.0	0.0	5.5	0.0	61.2	0.0	0.0	0.0	0.0	0.0	0.0	4.6	2.2	0.1	0.0	7.2	7.4	0.2	0.4	10.9
3H	0.7	1.9	0.1	0.2	94.5	0.1	0.3	0.4	0.1	0.0	0.1	1.3	0.0	0.0	0.0	0.0	0.0	0.1	0.0	0.0	0.1
59Ni	0.0	0.0	0.9	0.7	3.5	82.8	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	3.3	3.4	0.1	0.2	5.0
60Co	0.0	0.0	0.5	12.1	17.8	58.1	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	0.1	0.0	3.1	3.2	0.1	0.2	4.7
63Ni	4.5	46.3	1.5	8.2	3.0	5.4	24.7	0.2	0.1	0.0	0.3	2.9	0.1	0.0	0.0	0.0	0.8	0.8	0.0	0.0	1.2
79Se	14.2	7.4	0.1	27.0	0.5	30.1	1.2	5.1	0.5	0.1	0.3	1.0	0.2	0.2	0.1	0.1	3.3	3.4	0.1	0.2	5.0
90Sr	4.0	0.2	3.0	2.2	12.7	19.8	3.7	0.6	1.0	1.2	17.9	28.4	0.1	0.1	0.1	0.0	1.4	1.4	0.0	0.1	2.1
93mNb	0.0	0.0	0.0	0.4	0.0	80.6	2.9	0.7	0.6	0.2	0.6	2.4	0.0	0.0	0.0	0.0	3.2	3.3	0.1	0.2	4.8
93Zr	0.0	0.0	0.0	0.5	0.0	82.2	1.8	1.0	0.7	0.2	0.4	1.4	0.0	0.0	0.0	0.0	3.3	3.3	0.1	0.2	4.9
99Tc	7.7	4.2	0.4	3.0	76.5	1.6	0.9	0.5	0.1	0.4	0.5	2.8	0.0	0.0	0.0	0.0	0.4	0.4	0.0	0.0	0.5

Table 3-10 Concentrations Compared to the Maximum Concentration in Each Component

	C-101	C-102	C-103	C-104	C-105	C-106	C-107	C-108	C-109	C-110	C-111	C-112	C-201	C-202	C-203	C-204	C-301	244-CR	DB	Pits	Pipelines
113mCd	0.0	0.2	0.8	3.5	6.4	100.0	0.0	0.1	0.1	0.0	0.0	0.0	0.5	0.5	0.5	0.4	10.6	10.6	12.9	12.5	10.6
126Sn	2.5	0.1	0.0	0.7	0.0	100.0	0.0	0.0	0.0	1.8	0.0	0.0	0.1	0.1	0.1	0.1	10.3	10.3	12.5	12.1	10.3
129I	7.3	7.9	15.8	3.4	100.0	3.0	4.1	0.1	0.2	1.7	0.0	0.1	0.0	0.6	1.5	0.0	2.6	2.6	3.2	3.1	2.6
137Cs	9.3	5.0	5.6	7.7	100.0	8.5	0.5	0.6	0.5	0.2	0.7	1.4	1.0	0.9	1.6	0.7	2.7	2.7	3.3	3.2	2.7
14C	1.7	7.5	6.8	4.0	100.0	7.3	7.9	5.9	0.9	1.8	1.9	4.4	11.8	3.1	3.1	3.5	4.8	4.8	5.8	5.7	4.8
151Sm	0.0	0.2	0.0	59.1	0.1	100.0	54.7	0.1	0.1	0.0	0.0	0.0	5.3	5.4	5.3	5.0	18.0	18.0	21.9	21.2	17.9
152Eu	0.0	0.0	0.0	2.6	0.0	100.0	0.0	0.0	0.0	0.0	0.0	0.0	1.8	1.9	1.8	1.7	11.0	11.0	13.3	12.9	10.9
154Eu	0.0	0.0	6.9	10.2	0.0	100.0	0.0	0.0	0.0	0.0	0.0	0.0	7.3	7.5	1.4	5.2	13.9	13.9	16.8	16.3	13.8
155Eu	0.0	0.0	6.3	4.4	0.0	100.0	0.0	0.0	0.0	0.0	0.0	0.0	3.3	3.4	5.0	3.1	12.5	12.5	15.2	14.7	12.5
226Ra	0.1	2.2	0.0	0.1	0.1	100.0	0.0	0.1	0.1	0.0	0.0	0.0	0.0	0.0	0.0	0.0	10.0	10.0	12.1	11.8	10.0
227Ac	0.1	1.3	0.0	0.9	0.1	100.0	0.4	0.0	0.3	0.1	0.2	0.4	0.0	0.0	0.0	0.0	10.1	10.1	12.3	11.9	10.1
228Ra	0.0	100.0	1.9	45.8	0.0	4.8	0.8	0.1	0.0	0.0	0.1	0.0	0.6	0.6	0.3	2.5	5.7	5.7	6.9	6.6	5.6
229Th	0.0	0.0	0.0	0.7	0.0	100.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	10.1	10.1	12.2	11.8	10.0
231Pa	0.0	0.0	0.0	4.3	0.1	100.0	0.6	1.0	1.1	0.3	0.3	0.5	0.0	0.0	0.0	0.0	10.7	10.7	12.9	12.5	10.6
232Th	0.0	51.5	4.0	100.0	0.0	10.4	0.4	0.2	0.0	0.0	0.0	0.0	1.3	1.3	0.7	5.5	12.4	12.3	15.0	14.5	12.3
232U	0.0	0.1	0.0	100.0	0.0	0.9	0.0	0.0	0.0	0.0	0.0	0.0	0.1	0.1	0.3	0.2	10.1	10.1	12.3	11.9	10.1
233U	0.0	10.2	0.2	100.0	0.0	0.1	0.0	0.0	0.0	0.0	0.0	0.4	0.0	0.0	0.0	0.0	10.0	10.1	12.2	11.8	10.0
234U	39.8	8.5	0.6	25.6	23.4	0.0	1.7	1.1	0.5	0.1	0.7	0.8	26.9	26.0	100.0	73.2	25.4	25.4	30.8	29.9	25.2
235U	43.0	8.6	0.8	28.7	23.7	0.0	1.8	1.5	0.6	0.1	0.5	6.2	25.7	24.7	100.0	71.4	25.4	25.4	30.7	29.7	25.2
236U	61.9	32.2	2.3	40.4	68.8	0.1	2.6	1.3	0.8	0.2	2.0	9.5	52.3	35.2	100.0	61.6	29.4	29.4	35.7	34.5	29.2
237Np	3.0	0.9	3.3	25.6	0.1	11.9	1.5	0.0	0.2	0.3	0.2	1.4	13.2	11.2	0.1	100.0	16.6	16.6	20.1	19.4	16.5
238Pu	2.2	7.2	18.4	11.1	22.6	30.8	0.4	0.0	0.3	0.3	0.6	1.3	100.0	90.3	3.7	0.1	25.5	25.5	31.0	29.9	25.4
238U	42.0	9.0	0.8	28.0	24.9	0.0	1.8	1.4	0.6	0.1	0.2	6.1	28.2	25.1	100.0	74.6	25.9	25.8	31.3	30.3	25.7
239Pu	3.9	23.5	2.0	2.7	44.5	6.0	1.6	0.2	0.2	0.6	0.5	0.6	100.0	90.5	3.7	0.1	20.6	20.6	24.9	24.1	20.4
240Pu	1.9	10.2	1.9	3.8	40.8	6.0	0.8	0.1	0.1	0.3	0.3	0.3	100.0	90.6	3.7	0.1	20.6	20.6	25.1	24.2	20.5
241Am	4.8	26.3	8.0	18.7	100.0	96.7	5.7	1.2	0.8	0.1	6.5	8.6	65.2	32.1	1.0	0.1	22.4	22.4	27.0	26.2	22.2
241Pu	3.2	15.1	1.3	11.4	27.9	12.6	0.3	0.0	0.5	0.3	0.1	2.0	100.0	90.0	3.7	0.1	22.1	22.1	26.6	25.8	21.9
242Pu	0.1	0.0	0.1	100.0	2.5	1.4	0.0	0.0	0.0	0.0	0.0	10.1	9.7	8.8	0.4	0.0	12.1	12.0	14.6	14.1	12.0
243Am	1.9	7.4	0.2	44.8	9.2	17.9	2.2	0.5	0.3	0.0	0.1	0.3	100.0	48.3	1.5	0.2	21.4	21.4	25.9	25.1	21.3
243Cm	0.0	0.0	0.0	9.6	0.0	100.0	0.0	0.0	0.0	0.0	0.0	0.0	97.7	47.3	1.5	0.1	25.6	25.6	31.0	30.0	25.4
244Cm	0.0	0.0	0.0	10.0	0.0	76.2	0.0	0.0	0.0	0.0	0.0	0.0	100.0	48.3	1.5	0.2	23.6	23.6	28.6	27.7	23.4
3H	0.2	0.9	0.0	0.1	100.0	0.0	0.0	0.2	0.1	0.0	0.0	0.2	0.0	0.0	0.0	0.0	0.1	0.1	0.1	0.1	0.1
59Ni	0.0	0.0	1.2	1.2	9.8	100.0	0.0	0.0	0.0	0.0	0.0	0.0	0.7	0.7	0.7	0.6	10.5	10.5	12.7	12.3	10.4
60Co	0.0	0.0	0.9	30.4	71.3	100.0	0.0	0.0	0.0	0.0	0.0	0.0	1.9	1.9	2.0	1.7	13.9	13.9	16.8	16.3	13.8
63Ni	5.3	100.0	3.5	25.0	14.5	11.3	14.7	0.4	0.2	0.1	0.3	1.7	2.5	0.6	0.2	0.1	4.4	4.4	5.3	5.2	4.4
79Se	20.1	19.3	0.2	100.0	2.8	76.7	0.9	10.6	1.6	0.4	0.4	0.7	7.7	7.9	7.7	7.2	22.1	22.0	26.7	25.8	21.8
90Sr	7.5	0.8	11.1	10.6	100.0	67.0	3.5	1.5	4.8	5.1	33.5	27.2	4.5	8.6	4.9	3.2	12.1	12.1	14.7	14.2	12.0
93mNb	0.0	0.0	0.0	0.8	0.1	100.0	1.0	0.7	1.1	0.3	0.4	0.8	0.2	0.2	0.2	0.2	10.4	10.4	12.5	12.1	10.3
93Zr	0.0	0.0	0.0	0.9	0.1	100.0	0.6	1.0	1.1	0.3	0.3	0.5	0.2	0.3	0.2	0.2	10.4	10.4	12.6	12.3	10.4
99Tc	2.4	2.4	0.3	2.4	100.0	0.9	0.1	0.2	0.1	0.3	0.2	0.4	0.3	0.2	0.3	0.4	0.5	0.5	0.6	0.6	0.5

Because many different wastes were sent into the system and transferred between components of the system, the uncertainty in using process knowledge and transfer-based modeling to estimate chemical and radiological compositions is large. Sending diverse waste streams to tanks resulted in complex physical, chemical, and thermal conditions. The volume of transfers in some tanks was quite large, such that small errors in information (volumes, phases, concentrations) could propagate into large differences over time. For example, Tank C-105 had a total traffic of 27,117 kgal (100,000 m^3) compared to a capacity of 530 kgal (2,000 m^3), or a total turnover of the volume of over 51 times. Considering that the total volume was unlikely to be removed and replaced at a time, the turnover rate of some layers or zones in the tanks could be effectively hundreds of times over. The use of modeling to estimate remaining waste phases and concentrations is a considerable engineering challenge.

The HDW model was used for many of the radionuclide inventories found in the BBI (RPP-19822). Staff reviewed the HDW model to understand how it works and if the inventory estimates are reliable. Staff requested the spreadsheets and background documents from DOE. Figure 3-18 provides a comparison of the predicted sludge concentrations from HDW 5.0, with the actual sample concentrations for select risk-significant radionuclides.

The modeled concentrations do not demonstrate any significant bias in deviations from the sampled values. The modeled concentrations are mostly within ± 2 orders of magnitude with the majority within ± 1 order of magnitude. The uncertainties in the modeled results with HDW are significantly larger than those applied to the overall radionuclide inventory in the PA which is generally ± a factor of 2 for most radionuclides. The HDW model was revised in 2004 from version 4.0 to version 5.0 (RPP-19822). The revisions encompassed three main areas: updated ORIGEN2 fuel activity estimates, improved chemical process simulation, and general error correction. Some of these changes resulted in substantial changes to estimated inventories including for key radionuclides. Modeling of process losses resulted in decreases for ^{14}C and ^{99}Tc of 82 and 23 percent respectively. Numerous significant changes were the result of error correction. Correction of material balance errors increased Ni isotope concentrations by over 40%. A change to the half-life of ^{79}Se resulted in the inventory to decrease by 90 percent. Incorrect entry of a fuel activity in HDW 4.0 resulted in a 352 percent increase in projected ^{243}Am activity. Even with version 5.0 the total HDW model was not checked (legacy calculations). Even with these changes the differences between simulated and observed concentrations were significant. Given the QA issues and the observed differences, 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 (Recommendation #23). In general, modeling, process knowledge, waste templates, and other subjective methods are unlikely to be sufficiently reliable to provide inventory estimates for risk-significant radionuclides given the complexity of tank farm operations and processes. These methods are likely useful for planning and screening purposes, but final inventory estimates should be based on sampling.

DOE did not clearly distinguish which radionuclide inventory results were based on sampling and which were based on other methods such as modeling or process knowledge. It is recommended in future waste determinations that DOE provide more transparency as to the source of the inventory information such that the assigned uncertainty ranges can be better understood and evaluated (Recommendation #24).

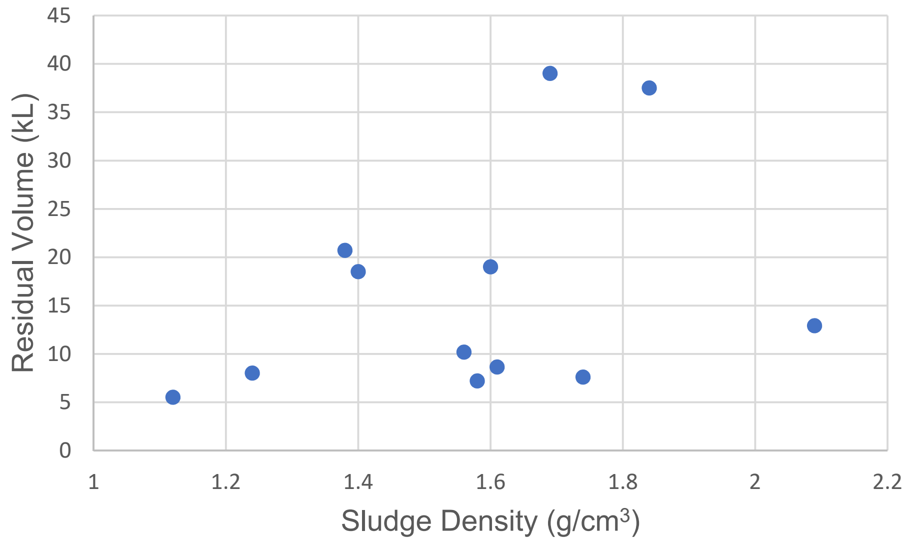


Figure 3-17 Comparison of Residual Waste Volume and Sludge Density

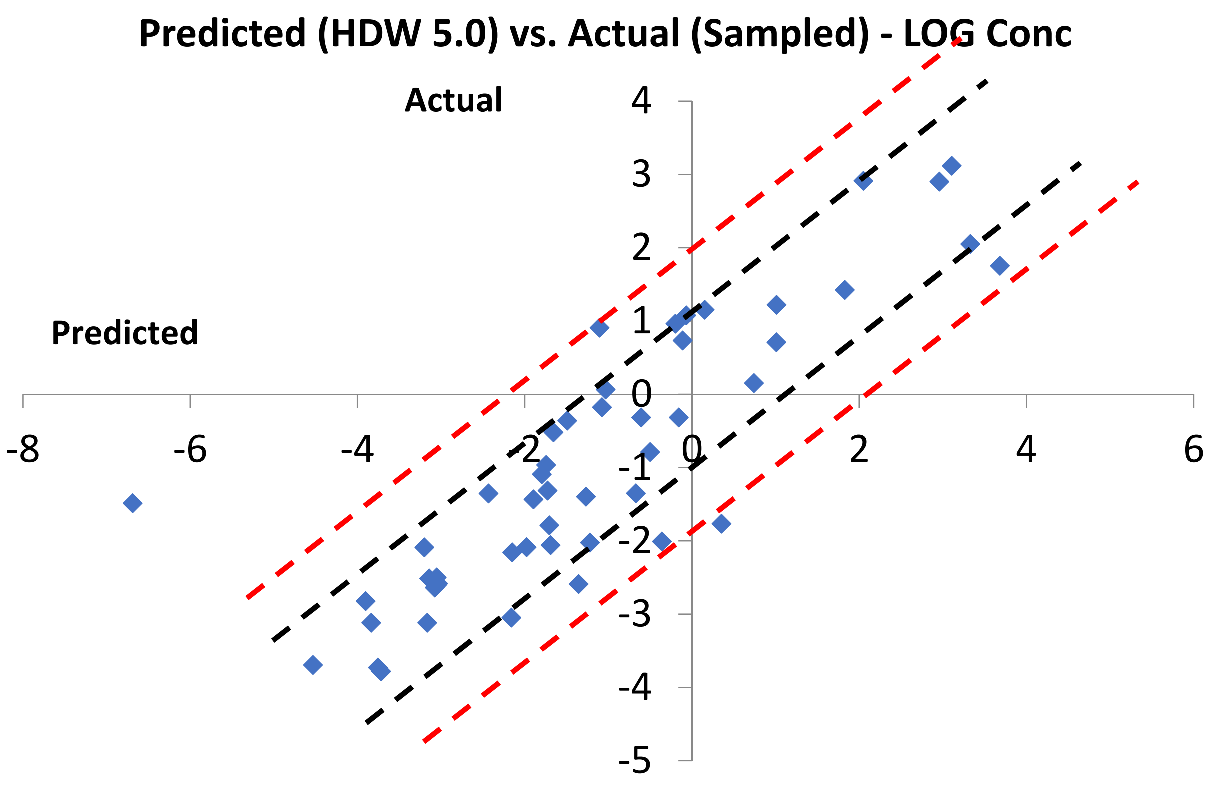


Figure 3-18 Comparison of HDW 5.0 Radionuclide Concentrations in Sludge with Results from Sampling

The CCMS approach to estimating residual volume after retrieval is an appropriate method and yields more reliable results than previous methods. Continued use of trained operators will help maintain the accuracy of the results. The uncertainty in the volume remaining is typically much lower than the uncertainty in the concentrations of key radionuclides.

Overall, the NRC staff did not identify biases in the inventory estimates that would change the projected inventory such that the conclusions of the Draft WIR Evaluation with respect to tank closure would be invalidated. However, the uncertainty in the residual inventory has likely been underestimated because some sources of uncertainty were not included. In addition, the statistical methods used may have understated the uncertainty in tank inventory. It is recommended that uncertainty in the radiological inventory be expanded and uncertainty in analytical methods should be included in future evaluations (Recommendation #25).

3.5.2.2 NRC Evaluation of Residual Ancillary Equipment Waste Inventory

NRC reviewed DOE's estimates of inventory assigned to ancillary equipment including catch tank C-301, the 244-CR vault, pump pits, diversion boxes, and pipelines.

The inventory estimate for the C-301 catch tank was based on measurement of the levels of liquid and sludge remaining in the systems. DOE assumed that 90% of the waste would be retrieved prior to closure. NRC staff finds this is a reasonable assumption given the geometry of these systems (relatively small) and that access may be less limited than with the larger waste tanks. The radionuclide concentrations prescribed for the waste was based on the average measured concentration of all waste remaining in WMA C. NRC does not believe, given the unique operating history of the C-301 catch tank that the average tank farm radionuclide concentrations will be accurate estimate for actual concentrations. DOE indicated in response to the NRC's RAI that a data quality objectives report and sampling analysis plan were being prepared to characterize the contents of the C-301 catch tank, and that the waste classification of the tank and the performance assessment calculations for the tank will be revised after characterization is completed. NRC agrees that sampling of the tank is the proper action to address the uncertainty in the radionuclide concentrations.

The 244-CR Vault contains four tanks (001 and 011 are 50,000 gal (189 m^3) and 002 and 003 are 15,000 gal (57 m^3). The 244-CR Vault was used for a larger variety of different applications, most importantly the precipitation of ^{137}Cs, ^{90}Sr, and ^{60}Co from TBP wastes. The radionuclide concentrations prescribed by DOE for the waste in the 244-CR Vault was based on the average measured concentration of all waste remaining in WMA C. Use of the tanks for precipitation has the potential to produce sludge with significantly higher concentrations than the average concentrations in WMA C. NRC had raised the issue of the inventory in the 244-CR Vault in the NRC RAI. NRC had also raised an issue about the fact that DOE had distributed the inventory over the plan view surface area in the vault rather than looking at the waste in individual tanks. In response to the RAI, DOE evaluated intrusion into tank CR-011 and estimated an acute intruder dose of 0.44 mSv (44.2 mrem) compared to 0.039 mSv (3.9 mrem) in their base case. That intruder dose would be the largest from any component within WMA C and when combined with higher concentrations in the residual waste would be a risk significant technical issue identified in the NRC review. DOE indicated in response to the NRC's RAI that a data quality objectives report and sampling analysis plan were being prepared to characterize the contents of the 244-CR Vault, and that the waste classification of the vault and the performance assessment calculations for the vault will be revised after characterization is completed. NRC agrees that sampling of the vault is the proper action to address the uncertainty in the radionuclide concentrations. Video inspection of the 244-CR Vault identified potential in-leakage and possible intra-vault leakage that contributed to high liquid levels within the vault (RPP-RPT-24257). The report RPP-PLAN-47559 noted that on February 18, 1965, the 244-CR Vault was found flooded up to the level of the tank tops. Whereas it does not appear that waste was released from the tanks into the vault, characterization of the material within the vault should be completed when the tanks within the vault are characterized.

For the waste transfer pipelines, pits, and diversion boxes both the volumes and concentrations were unknown. The residual waste in the pits and diversion boxes were expected to be small. DOE assumed that the pits and diversion boxes (7 total) either have been or will be flushed leaving only a thin layer (0.04 cm [0.1 inch]) of material over the surface area. The total waste in the pits was estimated to be 30 gal (0.11 m^3). The total waste remaining in the diversion boxes was estimated to be 60 gal (0.22 m^3). NRC agrees that the volume of waste in the pits and diversion boxes is likely to be small. However, visual inspection is unlikely to provide accurate estimates of the radioactivity present in these systems. Basic sampling or radiation detection measurements may be useful to verify the assumptions about waste remaining in these systems.

One of the most challenging aspects of closing WMA C may be the pipelines. There is approximately 11 km (7 miles) of pipelines within WMA C with different diameters and lengths of different segments (RPP-PLAN-47559). The pipelines are diverse, have a complex operational history, and have limited documentation. DOE's analysis of the pipelines in WMA C is almost completely assumption-based in terms of both the volume of waste and its radiological composition. DOE provided basis for the assumptions which were characterized as being conservative. DOE indicated that after usage the pipelines were routinely flushed to remove residual waste. Routine flushing would likely decrease the amount of waste that may remain in lines. However, some pipelines were known to have plugged completely and the operational history reports document numerous occurrences of plugging and unplugging. A plug can form from a discrete process or can result from a gradual build-up of material over time. Without monitoring pressure drops over time, the extent of the pipeline area that is filled with waste is unknown and cannot be inferred from the operational history that waste can be sent through the pipelines. The DOE assumption that the pipelines are 5% full of waste at the average concentration of radionuclides within WMA C may be reasonable or even pessimistic, but the assumptions are highly uncertain. Even if the pipelines were 100% full of waste at average concentrations, the PA demonstrated that the performance objectives would still be met. Because DOE is going to be closing many more facilities in the future, it is recommended that characterization of some sections of pipelines be completed to determine the quantity and composition of waste remaining to provide basis for future waste evaluations. The assumptions about the quantity and composition of waste remaining in unplugged pipelines are uncertain and could be quantified to better support future decision making throughout the Hanford Site for future evaluations (Recommendation #26).

The feasibility evaluation for pipelines in WMA C (RPP-PLAN-47559) describes pipelines that have failed. A failed pipeline from C-112 to the 252-C diversion box was identified as V172. The report states that the failed pipeline was isolated. The report also indicates that a line adjacent to the 152-A Diversion Box had a leak and was abandoned. Pipeline V-103 was abandoned due to pipeline leakage. It does not seem practical that failed and abandoned pipelines were flushed after they failed and therefore were isolated or abandoned. An isolated line was capped or otherwise blocked from future use. Abandoned lines were replaced and left in place. This report also indicated that various pipelines were installed and abandoned depending on the waste campaign and required routings. It was concluded that the waste composition in abandoned piping cannot be made based on specific transfer records of historical process records. NRC would agree with these conclusions. DOE's assumption of 5% waste remaining in failed and abandoned pipelines is not reasonable without additional technical basis.

One pipeline was known to have plugged (V122) and DOE assumed it to be 100% full of waste. For the purposes of the PA, DOE also assumed that the gravity-drained cascade lines that had a history of plugging and unplugging were 100% full of waste. The composition of waste in the pipelines was assumed to be equal to the average of all waste in WMA C. NRC questioned the assumed composition of waste in the plugged pipelines in the RAI. In response, DOE indicated that the most common waste types were Uranium or TBP waste and PUREX coating wastes, and each of these waste types had lower concentrations of the radionuclides that drive the impacts to inadvertent intruders. For the V-122 line, DOE indicated that the waste type was PUREX Supernatant (PSN) of PUREX high level waste (P2). They also indicated that additional records suggested that the line was likely unplugged because additional transfers occurred after the date of recorded plugging. Though it is important to understand the general waste types that were being transferred, the phenomena of plugging could be the result of solidification of the fluid being transferred or it could be the result of precipitation/deposition of solids. The solids fractions of waste being transferred could range from 0 on the low end to approximately 25% based on records (RPP-25113). Considerable increases in radionuclide concentrations can occur in the solids if the material in the plugged pipelines is due to precipitation or deposition, which is why the sludge in the tanks tends to have higher concentrations than the liquid phase in some cases by many orders of magnitude. The assumptions about the quantity and composition of waste in plugged pipelines are extremely uncertain and do not support the inventory values assigned for the performance assessment.

Many of the pipelines within WMA C are contained within encasements or lie upon support structures. The geometry of the encasements is quite variable. RPP-PLAN-47559 notes that some encasements can be quite large. The 200-series encasements are 61 cm by 61 cm [24 in by 24 in] or 61 cm by 31 cm [24 in by 12 in]. The 100-series tanks are 36 cm by 25 cm [14 in by 10 in]. The void space within encasements can be considerably larger than the void space within the nominal 7.6 cm [3 in] diameter pipe used in the base case intruder assessment (~50 cm^2 compared to up to ~3500 cm^2). Thickness of waste is an important factor in the inadvertent intruder assessment and in waste classification calculation. The same report assumed that encased pipes would be more difficult to remove because of the potential for encountering a contaminated environment within the encasement. Records indicate that leaks have occurred within encasements. Report RPP-RPT-29191 noted a leak that transported waste down an encasement, which eventually entered C-101, C-102, and C-103. NRC inquired about pipe encasements in the NRC RAI. DOE indicated that encasements had drains that connected them to pump pits, which drain into the tanks. The only known instance of a leak into an encasement was the report noted above. DOE concluded that, because the liquid levels in the tanks changed, there would be minimal waste left in the encasement. In addition, the waste that was being transferred at the time was low in concentrations of radionuclides that drive intruder doses compared to the average waste.

It appears that the only reason this leak was identified by DOE is because it involved a large loss of material, large enough to change the liquid levels in 100-series tanks by a noticeable amount (Note: The tanks experience condensation, evaporation, and in-leakage such that significant volumes of liquids are necessary to change tank levels by an observable amount (e.g., an addition of 3800 L [1000 gal] would only change the level of the tank by less than 1 cm [0.3 in])). Considering the number of unplanned releases that occurred over the history of WMA C operations, it is expected that numerous other leaks to encasements have likely occurred. Most waste that was transferred had solid fractions from 5% to 25%. Waste that leaked into encasements would not be subject to the pressure forces within the transfer line, such that solids would likely be deposited while liquids could freely drain back to the tanks once the driving force from the pressure was lost and the material was moving solely as a result of gravity. As discussed previously, DOE's records describe numerous pipelines that failed or were taken out of service. DOE has taken the most optimistic position that no other leaks to encasements have occurred to the numerous miles of pipelines operated for the past more than 50 years and if they did leak, no waste remains in the encasements. It was not within WMA C, but incident 72-26 described a waste line that failed, and the encasement drains became plugged, which is the scenario that could lead to significantly more inventory in the piping system than assumed by DOE (WHC-SD-WM-TI-773). Report RPP-RPT-46879 Rev. 0 noted that the three largest unplanned releases within WMA C were all due to failed pipelines, and indicated it was possible there were other unknown leaks in WMA C. It is recommended that DOE perform characterization of encasements to determine how much inventory is present (Recommendation #27).

If the impacts to a potential intruder are deemed to be low, it is not clear why the impacts to workers (and the expense for removal of a plugged line) would be extremely high as indicated by DOE in their response to the NRC RAI (DOE, 2019). Because they are potentially accessible much earlier, plugged, failed, and abandoned pipelines and their encasements arguably pose a higher risk to an inadvertent intruder compared to the residual waste in the large buried tanks with waste covered by many meters of grout and reinforced concrete. In addition, these systems are not characterized, and the volume and composition of waste is assumed, whereas the tanks are sampled and characterized after retrieval. Sampling and characterization of some of these systems may reduce uncertainties in the inventory estimates.

3.5.2.3 NRC Evaluation of Source Term Release

DOE's waste release modeling was limited to uranium and ^{99}Tc and was based on empirical observations supported by geochemical modeling. In the case of residual waste in WMA C, the residual waste is not an engineered wasteform and therefore most radionuclides were assumed to be available for release. This is a conservative approach and is acceptable, though significant performance improvement may be gained with additional study into waste release rates.

DOE completed empirical measurements to characterize and study the release of radionuclides from samples of waste obtained from tanks C-103, C-106, C-202, and C-203. The studies used techniques, such as XRD and SEM, to identify phases present and the changes to those phases in leaching experiments. The models that DOE developed from the empirical measurements were clearly described, consistent with the underlying technical reports, and were implemented as described in the performance assessment model. NRC staff did not identify any errors or inconsistencies. DOE provided comparisons between the models that were developed and the experimental data, which is very useful for justifying the models to external stakeholders. Comparisons between data, process models, and the PA model in the area of source-term release implementation is a good practice that should be implemented more regularly in future PAs (Recommendation #28).

The release modeling of ^{99}Tc is unlikely to be risk-significant under most scenarios. Given the time constants and initial release fraction most of the ^{99}Tc should be available for diffusion or advection in a relatively short amount of time compared to the timescales of the performance assessment. DOE's modeling of uranium release using a step-function for solubility was appropriate and consistent with the data. Though considerable complexity in geochemical modeling could be pursued, it is probably not warranted in the WMA C PA given the risk-significance and available supporting information.

DOE's determination of diffusion coefficients for the basemat was based on experiments discussed in Section 3.4.2.2.1. The effective diffusion coefficients include the effects of sorption because ^{99}Tc was used rather than a conservative species (i.e., one that does not experience sorption). The modeling of release in the PA applied the effective diffusion coefficient that included the effect of sorption and then also applied a distribution coefficient to the basemat concrete. Staff were able to verify that the flux rates were decreased when both were included. DOE should ensure that the effective diffusion coefficient values do not include sorption if it is going to be applied to the basemat layer in the model (Recommendation #29).

As the authors of PNNL-16748 cautioned, the release data consisted of empirical measurements that were the result of the phases present and the chemical conditions for release in the tests. The assumption that the observed data is representative and that the observed ranges will bound the results from other tanks is uncertain. This is not risk-significant for the WMA C PA but should be considered in future PA's. In general, uncertainty ranges for the PA should not be limited to the range observed from limited samples. Additional sampling would be expected to generate results that are outside of the range of the results from limited samples. If future empirical testing is performed for waste release, organic compounds present in tank residuals should be included as part of the experimental design (Recommendation #30).

Overall the waste release characterization and release modeling were high-quality and clearly documented. NRC discussed transport properties of the basemat and conceptual models for near-field advective transport in Section 3.4.2.2.1.

3.5.2.4 NRC Evaluation of Gaseous Near Field Transport

NRC staff reviewed DOE's modeling of gaseous near field transport. DOE considered four radionuclides that could potentially be transport in a gaseous form: ^{14}C as CO_2 gas, ^{3}H, ^{129}I, and ^{222}Rn. These radionuclides were appropriate to consider for potential gaseous transport.

The Henry's law constants DOE assigned, used to represent air-to-water partitioning, were large to conservatively overestimate the fluxes of gaseous species to the environment. NRC staff examined the total amounts transported by air and water of each of the four species discussed above in DOE's PA model. Staff also examined the timing of releases. For ^{3}H, it is conservative to maximize the air pathway releases because most of the ^{3}H will decay during transport in water prior to reaching a receptor access point. The peak tritium release to the air occurs in under ten years, which is non-physical. The steel liners are likely to be intact. Tritium is commonly seen in the water phase at disposal sites. If DOE uses similar models to plan for soil remediation the amount of tritium in soil may be underestimated because it has been simulated as being released to the air pathway. For ^{129}I, about eight times more ends up being transported through the air pathway than the water pathway. It is appropriate to maximize the transport through a pathway to perform screening calculations or even to conservatively demonstrate compliance with criteria. However, ^{129}I can be an important contributor to water pathways in most performance assessments. Henry's Law constants should be set to expected values in base case calculations for the water pathway (Recommendation #31). The transport of ^{14}C can be slower than ^{129}I, but the minimum values of the Kd distributions for the unsaturated zone, or UZ, are both 0 ml/g. Uncertainty cases may be underestimating the impacts from these radionuclides because they have been maximized for the air pathway.

DOE's approach of combining the fluxes from multiple sources to a single point source is appropriate to account for the impact from multiple sources to a receptor. The modeling of gaseous releases should examine shorter transport pathways if in fact the grout cannot be designed for zero shrinkage (Recommendation #32). A shrinkage gap between the tank wall and grout would allow diffusion in air up to and out of the inlet ports or other openings that are the sources of water flow into the tanks. Otherwise, DOE's calculations for diffusive transport of gaseous species were clearly described and appropriate. Once the fluxes reach the land surface, DOE used standard approaches (Gaussian plume) to estimate transport to a distant receptor. For the onsite receptor (intruder), DOE did not include gaseous releases. Gaseous releases should be included for the inadvertent intruder, especially from radon which is very sensitive to transport distances owing to its short half-life. Typically for outdoor exposures a simple box model is used with average annual wind speed. Because the fluxes released are slow and continual and the wind-speed distribution is highly-variable including periods of relative calm in most 24-hour periods, a simple box model approach will significantly underestimate onsite doses. A convolution of daily and annual wind speeds is needed to get an effective annual concentration in the air. For radon, emanation coefficients can be highly variable. If radon is found to be an important contributor to air pathway doses, DOE may want to consider including uncertainty in emanation coefficients.

Summary of Review

- The NRC staff reviewed DOE's development of the radionuclide inventory, source term release, and near field transport. DOE's approach to development of residual inventory for the tanks emphasized the use of post-retrieval sampling, when available. This is appropriate and likely to yield the results with the lowest uncertainty of the methods available. Overall, the NRC staff did not identify biases in the inventory estimates for tanks that would change the projected inventory such that the conclusions of the Draft WIR Evaluation for WMA C with respect to tank closure would be invalidated.

- Overall, the waste release characterization and release modeling were high-quality and clearly documented.

- DOE's assumption of 5% waste remaining in failed and abandoned pipelines is not reasonable without additional technical basis. The assumptions about the quantity and composition of waste remaining in unplugged pipelines are uncertain and could be quantified to better support future decision making throughout the Hanford Site.

- The assumptions about the quantity and composition of waste in plugged pipelines are extremely uncertain and do not support the inventory values assigned for the performance assessment.

- The uncertainty in the residual inventory has likely been underestimated because some sources of uncertainty were not included. In addition, the statistical methods used may have understated the uncertainty in tank inventory.

- Recommendations #23 through #32 with this section can found in Table 5-1 of this report.

3.6 Flow and Transport in the Unsaturated Zone

Flow and transport in the unsaturated zone are important processes at the Hanford Site because the unsaturated zone is approximately 80 m (260 ft) thick. The unsaturated zone can significantly delay the release of contaminants from reaching the aquifer, and eventually, an offsite receptor. Please refer to figures, tables, and descriptions in Section 1.2 on the geography, geology, and hydrogeology of the Hanford Site and WMA C.

3.6.1 Summary of DOE Analyses of Flow and Transport in the Unsaturated Zone

In the area of WMA C, the hydrogeological system consists of a thick vadose or unsaturated zone, roughly 70 m to 90 m (230 ft to 295 ft) thick. In the immediate vicinity and surrounding the actual tanks of the C-Tank Farm is a man-made mixture of backfill. This is a gravel-dominated mixture consisting of poorly to moderately sorted cobbles, pebbles, and coarse to medium sand. The thickness of the backfill is about 10 m (33 ft) with occasional layers of sand to silty sand occurring near the base of the backfill. Below the backfill is H1 Hanford formation, a less than 10 to 30 m (33 to 98 ft) thick gravel-dominated flood sequence composed of mostly poorly-sorted, basaltic, sandy gravel to silty sandy gravel. After that is the H2 Hanford formation, a 45 to 70 m (150 to 230 ft) thick sand-dominated flood sequence composed of mostly horizontal to tabular cross-bedded sand to gravelly sand. Some of the sand beds are capped with thin layers of silty sand to sandy silt. The undifferentiated H3/CCU/RF unit follows below. WMA C lies along the edge of a paleochannel where original deposits of H3 Hanford gravels, the Cold Creek unit, and the Ringold formation were eroded. Because of the difficulty in distinguishing between the reworked units, the lower sedimentary unit is identified as the undifferentiated H3/CCU/HF unit.

The water table below the WMA C lies within the H3/CCU/HF unit so that the upper part is unsaturated while the bottom part is saturated. The total thickness of this unit is less than 27 m (90 ft) with this thickness being roughly equally divided between the unsaturated zone and the aquifer below (RPP-RPT-46088, 2016). The H3/CCU/RF unit consists predominantly of sandy pebble- to cobble-sized gravel with occasional boulders. Mineralogically, the sand fraction consists of 15 to 60% basalt grains with generally less than 1 wt% calcium carbonate.

Clastic dikes are found in the Hanford formation. They are vertical to sub-horizontal fissures of various sizes filled by multiple layers of unconsolidated sand, silt, clay, and minor gravel aligned parallel to sub-parallel to dike walls. Thicknesses of the clastic dikes can be from 0.001 to 1.8 m (0.003 to 5.9 ft) thick while the depths of these features range from 0.3 to 55 m (1 to 180 ft). On the surface they can extend up to 100 m (328 ft) long. Clastic dikes have been observed in the Waste Treatment Plant (WTP) excavations and in the Integrated Disposal Facility (IDF) excavation (RPP-RPT-46088, 2016). Where the dikes intersect the ground surface, a feature known as patterned ground is observed. Patterned ground features are most abundant when Hanford formation sand-dominated and silt-dominated facies are at or near ground surface. No clastic dikes have been observed in WMA C. This is probably because sediment at WMA C is reworked Ringold sediment and course-grained sediment of the Hanford Formation (RPP-RPT-46088, 2016).

The hydrogeological system at WMA C is a significant barrier limiting the impact of releases from WMA C and is, therefore, a major component of the WMA C PA. DOE used the WMA C flow and transport model to simulate unsaturated flow and contaminant transport of radionuclides in the unsaturated zone below the C-Tank Farm.

In the WMA C PA, DOE used a hybrid approach to obtain simulated results from the unsaturated zone which included both deterministic and probabilistic approaches. In the deterministic approach, the STOMP© simulator process-based code was used in the analysis of post-closure flow and transport in the unsaturated and saturated flow systems and to examine a range of model parameters through sensitivity analyses. Additional transport analyses were carried out using a probabilistic approach, where the GoldSim-based system-level code was used to perform uncertainty analyses and additional sensitivity analyses to support the basis for comparisons with performance objectives under DOE Order 435.1.

In Appendix H of the WMA C PA (see Table H-1), DOE specifically identified safety functions associated with the unsaturated zone that provide specific functions and are relevant to the performance of the facility. The performance of these safety functions may be related to the function’s ability to dilute the concentration of the radionuclides by spreading the contaminants by dispersion, to slow the transport of the contaminants to the aquifer, or by the ability of the various Hanford formation units to sorb certain radionuclides and slow their movement. This section and the following section will discuss and evaluate performance of the following safety functions listed in DOE’s WMA C PA (2016): VZ (vadose zone)1 – Thickness of the vadose zone (delay); VZ2 - Sorption on unsaturated zone soils (sorption); and VZ3 - Dispersion in unsaturated zone (dilution).

3.6.1.1 GoldSim One-Dimensional Probabilistic Model

The deterministic analyses DOE developed with STOMP© are augmented using probabilistic analyses with an abstracted model of the groundwater system using GoldSim whereby STOMP© flow fields are used as inputs to the probabilistic GoldSim-based model. The GoldSim system model relies on the flow-field related parameters extracted from the STOMP©-based model to provide moisture content, saturation, and Darcy flux input values at discrete points of a coarse discretization of the flow system.

Figure 3-19 shows representative hydrostratigraphic columns for the tanks at WMA C and are compared with the vertical discretizations of the GoldSim system-level model and the STOMP© process-level model. Nodes of the process-level model were used to represent the moisture content and Darcy flux for the grid cells in the system-level model (colored brown).

GoldSim was specifically designed for performing PA analyses and provides a platform for coupling the processes in the PA in a system-level model and the tools for propagation of the uncertainty (RPP-CALC-60449, 2016). GoldSim was the primary software used to perform the uncertainty analysis as documented in Section 8.1 of the WMA C PA (2016). Mass flux from the unsaturated zone was calculated for each source term (i.e., each single-shelled tank, the C-301 catch tank, the 244-CR vault, and pipelines) and transported vertically to the aquifer without lateral dispersion. For the unsaturated zone, DOE described how the one-dimensional GoldSim model obtains the flow field-related parameters such as moisture content, saturation, and Darcy flux from the STOMP© output (RPP-RPT-58948, 2016). These flow field-related parameters were extracted from the three-dimensional flow and transport model as were the thicknesses of the Hanford formation units. Of special interest were the thicknesses under each of the twelve 100-series tanks and the four 200-series tanks.

DOE reported that the median value of the 100-series tanks is very similar to the thickness below tank C-105, and that the median value for the 200-series tanks is close to the thickness below tank C-203 (RPP-RPT-58948, 2016). Tank C-105 was selected as the representative column for the 23-m (75-ft) diameter (100-series tanks) flow-field abstraction and tank C-203 was selected as the representative column for the 6-m (20-ft) diameter (200-series tanks) flow-field abstraction.

For the base case calculations, DOE varied the recharge rates spatially and temporally within the STOMP model domain (see Table 3-2). Due to the presence or absence of the engineered surface cover and the degree of performance of the surface cover between 2020 and 2520, the moisture content and Darcy flux profiles within the soil column varied with depth and time. Figure 3-20 shows the vertical Darcy velocity or specific discharge for Tank C-105 over time, including a large anthropogenic recharge prior to the surface barrier in 2020.

The H2 Hanford sand unit is the thickest unsaturated zone unit in the vertical profile shown in Figure 3-19 and it is discretized into a 3.75-m (12.3 ft) grid cell at the top and 5-m (16-ft) grid cell at the bottom while the middle 40 m (130 ft) is discretized into 80 grid cells resulting in 0.5 m (1.6 ft) length cells. The flow field applied to pipeline releases was calculated separately. Vertical Darcy fluxes and volumetric moisture contents from the STOMP nodes that fall within the 150-m (490-ft) square pipeline source area but outside the tank footprint were averaged to calculate the pipeline flow field using the model layer thicknesses underneath the 100-series tanks. Advective flow occurred through the pipelines for all time periods and the immediate surroundings of the pipelines were modeled using hydraulic property values of the soil backfill.

3.6.1.2 STOMP Three-Dimensional Flow and Transport Model

The STOMP three-dimensional flow and transport model was used to estimate the future flow velocity in the unsaturated zone and concentrations of radionuclides and non-radiological contaminants entering the saturated zone to assist in evaluating the potential long-term safety impact from residual waste left in tanks and ancillary equipment at the WMMA C.

Hydrostratigraphy Below C-105	STOMP Node Elevation (m)	STOMP Node Numbers	STOMP Vertical Discretization (m)	GoldSim Vertical Discretization (m)
H1 Gravelly Sand	185.75	**69**	1.00	2 m
	184.75	68	1.00	
	183.75	67	1.00	3.5 m
	182.625	**66**	1.25	
	181.375	65	1.25	
	180.125	64	1.25	3.75 m
	178.875	**63**	1.25	
	177.625	62	1.25	
H2 Sand	176.375	**61**	1.25	3.75 m
	175.125	60	1.25	
	173.875	59	1.25	
	172.625	58	1.25	2 m (over 40 m thickness)
	171.375	57	1.25	
	170.125	56	1.25	
	168.875	55	1.25	
	167.625	54	1.25	
	166.375	53	1.25	
	165.125	52	1.25	
	163.875	51	1.25	
	162.625	50	1.25	
	161.375	49	1.25	
	160.125	48	1.25	
	158.875	**47**	1.25	
	157.625	46	1.25	
	156.375	45	1.25	
	155.125	44	1.25	
	153.875	43	1.25	
	152.625	42	1.25	
	151.375	41	1.25	
	150.125	40	1.25	
	148.875	39	1.25	
	147.625	38	1.25	
	146.375	37	1.25	
	145.125	36	1.25	
	143.875	35	1.25	
	142.625	34	1.25	
	141.375	33	1.25	
	140.125	32	1.25	
	138.875	31	1.25	
	137.625	30	1.25	
	136.375	29	1.25	
	135.125	28	1.25	
	133.875	27	1.25	
	132.75	26	1.00	5 m
	131.75	25	1.00	
	130.75	**24**	1.00	
	129.75	23	1.00	
	128.75	22	1.00	
H3 Gravelly Sand	127.75	21	1.00	7 m
	126.75	20	1.00	
	125.75	19	1.00	
	124.75	**18**	1.00	
	123.75	17	1.00	
	122.75	16	1.00	
	121.75	15	1.00	
	120.875	**14**	0.75	1.25 m
	120.25	13	0.50	
Aquifer	119.75	12	0.50	12 m
	119.25	11	0.50	
	118.625	10	0.75	
	117.75	9	1.00	
	116.625	8	1.25	
	115.25	7	1.50	
	113.75	6	1.50	
	112	5	2.00	
	109.5	4	3.00	
Inactive	106	3	4.00	
	102	2	4.00	
	97.5	1	5.00	

Node used in Flow-Field abstraction

Figure 3-19 Vertical Discretizations of the GoldSim System-level Model and the STOMP Process-level Model of the Hydrostratigraphy Beneath WMA C
[Figure 6-47 in DOE, 2016]

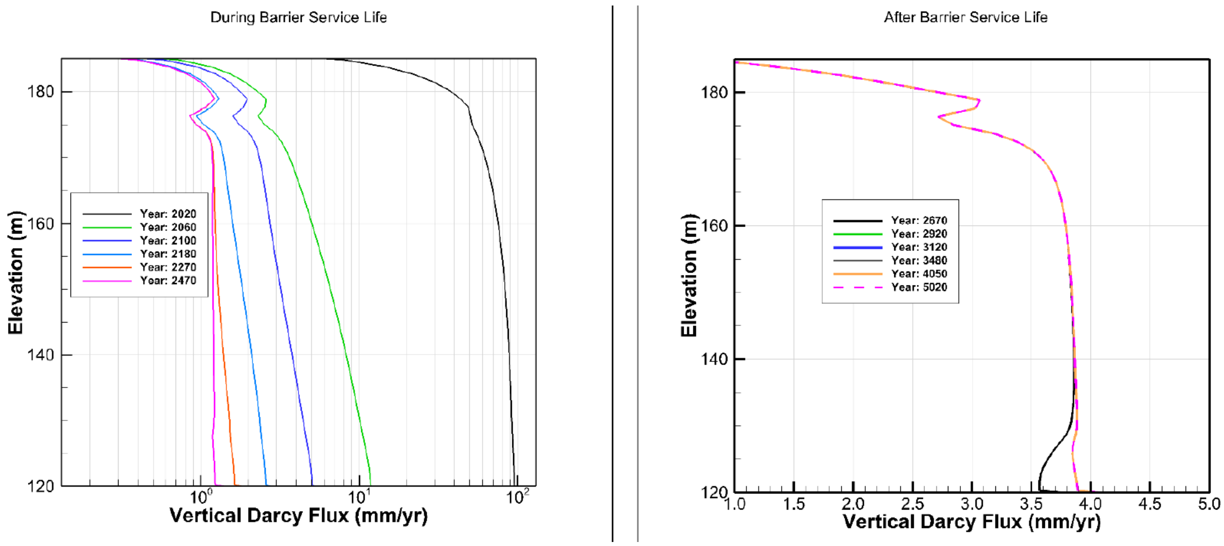


Figure 3-20 Vertical Darcy Flux Distribution in the Unsaturated Zone Below Tank 241-C-105 [Figure 4-4 in the RPP-RPT-58948 (2016)]

DOE documented the development of the numerical flow and transport model and the base case analysis (RPP-RPT-58949, 2016). In addition, the screening process used to identify and narrow the list of contaminants of potential concern that required evaluation in the STOMP model was also described. DOE discussed the parameters and model development necessary to review the base case and screening calculations performed (RPP-CALC-60448, 2016). The STOMP three-dimensional flow and transport model domain for the WMA C was rectangular in area with one side parallel to the general groundwater flow direction [length 738 m (2,420 ft)] and the other perpendicular to simulated groundwater flow [length 795 m (2,610 ft)]. The unsaturated and saturated zones in the base case were represented in the model by 104 m (341 ft) and 12 m (49 ft) respectively. The unsaturated portion of the STOMP flow and transport model was composed of (from the top down) backfill material, H1 gravel, H2 sand, and the undifferentiated H3/CCU/RF gravels. The Alternative Geologic Model II included an additional H2 gravel unit and H2 silty sand unit. The vertical base elevation of the model was represented by the basalt and the overall model thickness varied spatially according to the top of basalt elevation and surface relief.

A specified-flux boundary condition was applied at the top of the model and recharge rates depended on site and surface conditions simulated, the location and physical dimensions of WMA C, and the time of WMA C operations (Section 3.3 of this report discusses recharge). The bottom boundary of the unsaturated (vadose) zone was the water table and the bottom of the model (aquifer) was defined as a vertical no flow boundary condition (RPP-RPT-58949, 2016). No flow boundary conditions were also applied to unsaturated zone side boundaries and saturated zone boundaries running parallel to groundwater flow.

The flow and transport pathway process used for the WMA C unsaturated zone modeling was porous media continuum flow. The porous media continuum assumption and the soil moisture characteristics provided the basis for modeling the unsaturated zone. DOE described how the fluid transport characteristics associated with each geologic layer are obtained by approximating average upscaled values, with each unit having different flow and transport parameter values (hydraulic conductivity, bulk density, and dispersivity) (RPP-RPT-58949, 2016). Each heterogeneous formation was replaced by its homogeneous equivalent, and the upscaled or effective flow parameters of the WMA C PA were used to represent the equivalent homogeneous medium (EHM). Flow and transport models of the unsaturated zone required the specification of hydraulic properties for each discretized grid block (scales of the order of meters), which are much larger than the core scale at which the unsaturated properties are measured (WMA C PA, 2016). The process of defining large-scale properties for the numerical grid blocks based on small, core-scale measurements is called upscaling. However, the variability of field-measured moisture contents, induced by media heterogeneities, is inherently larger in comparison to the variability based on the flow and transport model simulations using the homogenized upscaled properties. Because of the upscaling, the output results show a general smoothing indicating that the field-scale variability is not being captured. Nevertheless, DOE indicated that this approach for developing the unsaturated zone hydraulic properties was adequate to approximate the flow and transport parameters and account for the significant heterogeneity present within the various geologic units (RPP-RPT-58949, 2016).

The basis for the unsaturated zone hydraulic properties (see Appendix B in the WMA C PA) for details on this process) began with the relatively extensive WMA C moisture content data collected since 2008. Overall, the moisture content data show considerable variability: the range of the measured data varied from a low of 0.11 (% volume) for backfill to as high as 30.64 (% volume) for H1 gravel-dominated unit. Soil-moisture and matric potential data for borehole samples inside and outside the C-Tank Farm were then compared by DOE. The averages for moisture content measurements for H2 sand inside the WMA C footprint with higher recharge were not significantly different from the region outside of the WMA C footprint with lower recharge. Subsequent evaluation of laboratory measurements for unsaturated zone soil moisture retention, saturated and unsaturated hydraulic conductivity for samples in the vicinity of WMA C and 200 Areas were then applied as the basis for the selection of hydraulic properties for the major hydrogeological units identified at WMA C. No site-specific hydraulic property data were available for WMA C. Instead, coarse sand hydraulic property data from nearby areas and WMA C moisture content distributions were used to identify and characterize hydraulic properties for the Hanford H2 sand unit. In appendix B in the WMA C PA DOE stated that the coarse sand unit of the Integrated Disposal Facility (IDF) site (Figure B-3) correlates well with the WMA C H2 sand unit. The data from the IDF was used as a surrogate for the WMA C H2 sands. The H2 sand sequence identified at the IDF site is ~61 m (~200 ft) thick and is the dominant facies at the site. No site-specific data was available for the WMA C H1 gravel-dominated unit and the undifferentiated H3/CCU/RF unit. However, as part of other Hanford Site projects, particle-size distribution, bulk density, saturated hydraulic conductivity, moisture retention, and unsaturated conductivity data have been collected for several borehole sediment samples (DOE, 2016) at other sites in the vicinity of WMA C and within 200 Areas.

DOE used selected properties to simulate an unsaturated zone flow field and the simulation results were cross-checked against WMA C field-measured moisture contents. A comparison of measured moisture profile for borehole C4297 and the simulated steady-state moisture profile for WMA C is shown in Figure 3-21. A simultaneous fit of both laboratory-measured moisture retention and unsaturated conductivity data was carried out. Van Genuchten-Mualem parameters for the various hydrogeological units were fit to the available data while the pore size distribution factor (ℓ) was kept fixed at 0.5.

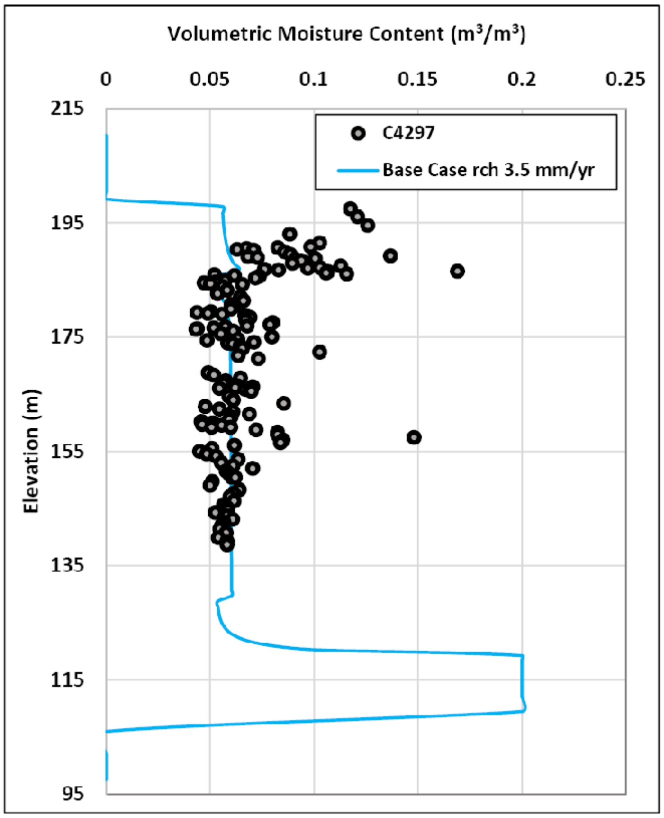


Figure 3-21 Comparison of a Measured Moisture Profile from a Borehole Within WMA C and a Simulated WMA C Steady-state Moisture Profile
[Figure B-14 in the DOE, 2016)]

Estimated unsaturated conductivities, based on the fitted saturated hydraulic conductivity (Ks) and the van Genuchten retention model parameters, have been shown to differ by up to several orders of magnitude with measured conductivities at the dry end. Table 3-11 gives a summary of the van Genuchten-Mualem parameters values and the characteristics associated with the unsaturated zone model layers of the process-level STOMP model for WMA C.

Upscaled flow parameters include moisture retention and hydraulic conductivity while upscaled transport parameters include diffusivity, sorption coefficients, macrodispersivity, and effective bulk density (used to calculate retardation factors for different species). Macrodispersivity reaches a constant value after the solute travels approximately 50 cm (~20 in) so that a longitudinal macrodispersivity value of 25 cm (10 in) was used for the H2 sand unit in the WMA C PA base case (in the STOMP model) based on the results of numerical simulation, stochastic theory and the 200 Areas experimental data (WMA C PA, 2016). For the gravel-dominant units (i.e., backfill, H1, and H3/CCU/RF) a longitudinal macrodispersivity value of 20 cm (8 in) was used. Transverse macrodispersivity was set at 1/10th of the longitudinal macrodispersivity. The tortuosity formulation in the Millington-Quirk model accounted for the ranges of moisture contents present in the unsaturated zone around WMA C and it was assumed by DOE that large-scale diffusion coefficients are a function of volumetric moisture content.

Table 3-11 Base Case van Genuchten-Mualem Parameter Values and STOMP Model Characteristics for the Unsaturated Zone at the WMA C

[Tables D-4 and 6-12 in DOE, 2016]

Strata	Number of Samples	θ_s	θ_r	α (1/cm)	n	ℓ^c	Fitted K_s (cm/s)
Backfill (Gravelly)	10	0.138	1.11×10^{-2}	0.021	1.374	0.5	5.60×10^{-4}
Hanford H1/H3 (Gravel-dominated)	15	0.171	1.11×10^{-2}	0.036	1.491	0.5	7.70×10^{-4}
Hanford H2 (Sand-dominated)	44	0.315	3.92×10^{-2}	0.063	2.047	0.5	4.15×10^{-3}
Vadose Zone Hydraulic Conductivity Anisotropy allowed to vary as a function of the moisture content.							
Vadose Zone Dispersivity Longitudinal to Transverse Anisotropy = 10:1							

The molecular diffusion coefficient for all species in pore water was assumed to be 2.5×10^{-5} cm^2/sec (3.9×10^{-6} in^2/sec) which was consistent with, and representative of, values used in other Hanford PAs (WHC-SD-WM-EE-004, 1995).

The geochemistry conceptual model component in the WMA C PA involves the partitioning behavior or sorption characteristics regarding release, retardation, and attenuation mechanisms and any simplifying assumptions for specific radionuclides contaminants (RPP-RPT-58949, 2016). The use of the constant soil adsorption coefficient, or adsorption-desorption distribution coefficient, (Kd) model was assumed to be generally applicable in the WMA C PA when contaminants are present at low concentrations, and the Kd values were chosen assuming low-salt, near-neutral waste chemistry (RPP-RPT-58949, 2016). The geochemistry conceptual models for the Hanford Site are based on laboratory studies, testing, and measurements of adsorption and desorption coefficients under saturated and unsaturated conditions involving Hanford Site-specific sediments, contaminants, and conditions (WMA C PA, 2016). DOE discussed how Kd values are typically lower for materials that contain significant amounts of gravel such as the backfill and the units of H1 gravel and H3/CCU/RF (PNNL-17154, 2008). DOE presented the basis for the Kd values, including gravel-corrected values, used to approximate the transport of the contaminants and radionuclides in Table 6-11 of the WMA C PA (2016).

A groundwater pathway screening analysis methodology was used to reduce the number of radionuclides that needed to be simulated in the PA modeling effort by distinguishing those contaminants and radionuclides that may impact groundwater during the specified compliance time period (i.e., 1000 years) and sensitivity and uncertainty analysis time period (i.e., 10,000 years). The STOMP model was used by DOE to determine the maximum Kd value of contaminants contained within the WMA C tank residuals that reach the water table within 1,000 and 10,000 years. Those with non-gravel-corrected Kd values less than 3 mL/g were included because the results of the screening analysis indicated that radionuclides with Kd values greater than 3 mL/g did not impact groundwater within the 10,000-year sensitivity-uncertainty timeframe.

Key assumptions made, and limitations for, the STOMP three-dimensional flow and transport model (WMA C PA, 2016; RPT-RPT-46088, 2016; RPP-RPT-58949, 2016) as they relate to the unsaturated zone included:

- Distribution coefficients (Kd) were used to represent sediment-contaminant chemical interaction that best represent plausible levels of reactivity. The Kd values were chosen assuming low-salt, near-neutral waste chemistry in the unsaturated and saturated zone.

- Applicability of model results was limited to radionuclides and non-radiological contaminants exhibiting linear isotherm behavior for contaminant release and attenuation, which neglect surface complexation and precipitation.

- Applicability of WMA C model results were limited to evaluations where hydrogeologic parameter values remain constant and unchanging over time.

- The unsaturated zone was modeled as an aqueous-gas porous media system where flow and transport through the gas phase was assumed to be negligible.

- Hydraulic property heterogeneity was assumed to be insignificant within geologic units, and each geologic unit within the unsaturated zone was assigned upscaled, effective hydraulic properties.

- When the contaminants of the source term were being transported to the saturated zone, the vertical mass transport in the unsaturated zone did not undergo lateral dispersion and stayed within the footprint of the source area.

An independent evaluation of the unsaturated zone conceptual model and the EHM approximation was performed using a moisture content database from a 200 East Area site that served as a proxy for WMA C (Ye at al., 2005). Insights were gained on large-scale moisture movement within a diverse heterogeneous media and a relatively dry moisture regime. Small-scale core measurements for hydraulic properties were used to predict the large-scale flow behavior at the 200 East Area site. A second approach was used by DOE that inverts the large-scale unsaturated properties using the temporal evolution of the moisture content distribution. For both the forward as well as the inverse approaches, simulated movements of the plume based on the effective hydraulic conductivities were claimed by DOE to be in good agreement with those for the observed plume.

In addition, two heterogeneous modeling approaches based on combining “soft” data (e.g., initial moisture content, bulk density and particle-size distribution) and “hard” data (e.g., soil hydraulic properties) (WMA C PA, 2016) were used by DOE. The use of both soft and hard data was valuable in reproducing the detailed moisture plume for the two heterogeneous models, and analyses were used to quantify the center of mass and the spread of the injected water for the observed and simulated moisture plumes. No significant differences were observed between these models and the EHM-based models and the heterogeneous models were able to reproduce the spatial and temporal behavior of the observed plume. Although the EHM-based modeling does not capture the detailed plume behavior, output from both the EHM and heterogeneous models were of similar magnitude. DOE used the results to justify use of the EHM approximation for unsaturated zone modeling.

Unsaturated zone modeling results were compared with measured data in the vicinity of WMA C. The intermediate calculations for simulated average moisture for different units were in overall agreement with field data and thereby increased confidence in the PA modeling approach (i.e., the EHM approach and calculations). However, the variability of field-measured moisture contents of the heterogeneous medium was larger in comparison to that based on PA simulations using homogenized upscaled properties, and the collective average, embedded in EHM approximation, cannot capture the field-scale variability (WMA C PA, 2016).

DOE performed a comparison to evaluate the unsaturated zone model developed for the WMA C PA against the TC&WM EIS model (DOE/EIS-0391, 2012) developed for the WMA C. No direct comparison of results was possible due to varying inputs related to the residual waste inventory and waste release processes. However, after making the STOMP WMA C model inputs consistent with the TC&WM EIS model, the flux at the base of the unsaturated zone was compared between the two models for ^{129}I. Results of the comparison, provided in Appendix G in WMA C PA (2016), indicated that the hydraulic properties used in the two models produced similar results.

3.6.2 NRC Evaluation of Flow and Transport in the Unsaturated Zone

In the WMA C PA, DOE performed correlation analysis between the input parameters and the peak dose to identify uncertain parameters that influence the magnitude of the peak dose regardless of the time. In this analysis, pore-water velocity in the unsaturated zone ranked as the second highest parameter after the saturated zone Darcy flux multiplier. Within 10,000 years, DOE identified pore-water velocity in the unsaturated zone as the most important barrier to affect peak dose. Within the 1000-year timeframe, the H2 Hanford sand distribution coefficient for Tc-99 had a correlation close to one and the pore-water velocity in the unsaturated zone ranked third. For the sensitivity analysis, sensitivity case VZP03 had parameter values set equal to the base case except the unsaturated zone hydraulic properties were changed to the 95th percentile values (see Section 8.1.4 and Table 8.7 in the WMA C PA). This sensitivity case showed an increase in maximum concentration at the POC of 60%. Due to these results and other similar analyses, the base case values associated with the unsaturated zone need to have technical bases, and analyses for sensitivity and importance of barriers need to encompass a sufficient range to bound uncertainty.

As previously discussed, the unsaturated zone in the WMA C PA base case has hydraulic properties where each heterogeneous geologic unit is replaced by an EHM with macroscopic flow properties. Small-scale laboratory measurements are upscaled and used to predict the large, field-scale flow behavior. DOE used the simulated flow fields to predict the mean flow behavior at the field-scale. However, during previous WMA C workshops between DOE, the State of Washington, and stakeholders, questions were raised as to the possible impact of heterogeneity in the unsaturated zone on contaminant transport. Consequently, DOE evaluated alternative heterogeneous models and tried to determine if representing heterogeneity at finer scales within an unsaturated modeled domain might produce results that are significantly different than those obtained for the PA base case analysis. These alternative hydrogeological conceptual models are discussed in Section 3.2, however will be reviewed in detail in the following sections.

Alternative Geologic Model I in the WMA C PA is the geological and hydrogeological conceptual model used in the base case. It was incorporated into the STOMP three-dimensional flow and transport model and interpolated onto the numerical grid. Alternative Geologic Model II was developed to address the questions raised about potential impacts of heterogeneity in the unsaturated zone on contaminant transport. The model was used to examine possible lateral flow and transport within the unsaturated zone, specifically in the H2 Hanford sand formation due to stratigraphic heterogeneity. Alternative Geologic Model II divided the Hanford H2 sand unit into three distinct subunits: the unaltered Hanford H2 sand, the more transmissive Hanford H2 gravelly sand, and the relatively thin, less transmissive Hanford H2 silty sand. The hydraulic properties associated with the 5^{th} and 95^{th} percentile realizations of unsaturated hydraulic conductivity curves developed for the Hanford H2 sand unit (see Table 8-7 in the WMA C PA) were considered representative of the Hanford H2 gravel/coarse sand and the Hanford H2 silty sand subunits, respectively.

Alternative Geologic Model II was used as one of the sensitivity cases in Section 8.2 of the WMA C PA and designated sensitivity case VZP04. The results of this alternative conceptual model of the hydrogeology indicate that the additional units with varying hydraulic properties do not strongly affect the maximum concentration results. Arrival times are 160 years earlier, most likely due to the inclusion of the gravelly H2 layer; however, the maximum concentration itself is about equal with the base case, 29 pCi/L (1073 Bq/m^3) compared to 30 pCi/L (1100 Bq/m^3). DOE reported similar results in RPT-CALC-60793 (2016). NRC staff asked DOE during a public teleconference (NRC, 2018b) what effect moving the H2 silty sand layer higher up in the soil column would have on lateral flow in the model. DOE stated that the effect of locating the silty layer elsewhere in the soil column would have a negligible effect on flow and resultant doses. DOE also stated that simulated lateral flow was not observed and that the lack of lateral flow was likely due to the generally dry subsurface conditions.

DOE completed sensitivity analysis to evaluate an additional geologic model, but not an alternative conceptual model since the probability of the additional feature (i.e., clastic dikes), existing in the subsurface of WMA C were believed to be sufficiently low that the alternative was excluded. However, stakeholder interest in an alternative that included a potential preferential pathway, such as a clastic dike or unsealed borehole located underneath Tank C105, was high. DOE included a “what if” alternative as a sensitivity case which featured clastic dikes and designated as sensitivity case VZP05. Sensitivity case VZP05 was identical to Alternative Geologic Model I, except that a clastic dike was assumed to exist under tanks C-102, C-105, C-108, and C-111, and another was assumed to exist under tanks C-110, C-111, and C-112. These clastic dikes started at the top of the H2 sand layer and extended vertically downward roughly 25 m (82 ft) in addition to extending across the WMA C. The hydraulic parameters assigned to the clastic dike material were selected to provide a preferential flow path for the residual waste (Section 8.2.3 in the WMA C PA). This evaluation of the alternative conceptual model of the geology indicated that the additional clastic dikes do not strongly affect the results of the WMA C PA. The peak concentration values showed little change from the base case value, although the clastic dike analysis showed a slightly reduced travel time (95 years).

Another alternative conceptual model used to identify the impact of heterogeneity for predicting concentrations in the water table and to compare with the base case was the heterogeneous alternative geologic conceptual model. Site studies of the 200 Area indicated that moisture content may be an indicator of sediment type and may impact subsurface transport (Ye and Khaleel, 2008). In addition, DOE stated that in-situ moisture measurements in a semi-arid to arid area would be anticipated to broadly correlate with sediment texture so that higher moisture contents are associated with fine-textured sediments and lower moisture contents are associated with coarse-textured sediments (RPP-CALC-60345, 2016). Consequently, this alternate unsaturated model development used the baseline soil moisture distribution from an extensive set of WMA C moisture content data, obtained from neutron moisture logging of direct push borehole and drywell locations, as an indicator of sediment texture. A geostatistical analysis of the moisture content database subsequently helped assign hydraulic properties for the heterogeneous media model (RPP-CALC-60345, 2016). A key assumption in all of this is that flow is in equilibrium with natural recharge.

Model runs were made, and contaminant breakthrough curves generated for the heterogeneous model. After DOE compared results to those based on the base case model, peak concentrations were found to be in good agreement between the two models, with the peak for the heterogeneous case was slightly lower than the base case. The calculated moisture contents of the alternative heterogeneous conceptual model were also generally in good agreement with field measurements. DOE discussed how heterogeneities induced by sediment texture and variability in moisture content lead to increased dispersion, resulting in a lower peak concentration for the heterogeneous model versus the base case model (RPP-CALC-60345, 2016). Thus, DOE concluded that the model results enhance the credibility of use of an EHM modeling approach for the WMA C PA.

DOE concentrated much effort to demonstrate that the WMA C PA can adequately simulate flow and transport in unsaturated zone. Stakeholder interest in the unsaturated zone's role in the transport of contaminants was also large. Any contaminants that are released near the surface will have to travel through a thick layer of unsaturated, unconsolidated material. The properties of this material will have a significant role in determining the speed of travel and concentration of those contaminants arriving at the water table. In addition, compared to the saturated zone, the unsaturated zone is more accessible, and much information has been gathered from the Hanford formation. Various stakeholders, such as the Nez Perce, have specific geologic interpretations that helped DOE develop additional alternative geologic models (WMA C PA, 2016).

The importance of infiltration and of recharge rates on the arrival time of the peak concentration at the point of calculation is demonstrated by the sensitivity case INF03 as discussed in Section 3.3.2.2 of this report. Peak concentration of ^{99}Tc is approximately 50% higher in sensitivity case INF03 than that of the base case; however, the arrival time after the surface cover degrades is more than an order of magnitude shorter than the base case arrival time, 74 years compared to 1055 years. For sensitivity case INF01, base case conditions are unchanged, except for a net infiltration rate of 5.2 mm/yr, or 0.20 in/yr (vs. 3.5 mm/yr, or 0.14 in/yr) that was applied to the surface cover area after a 500-year cover design life. Peak concentration increased, and travel time decreased, by roughly 20% for both (See Table 8-16 in the WMA C PA). To summarize, the concentration results of the base, INF02, and INF03 cases were 30, 36, and 47 pCi/L (1110, 1330, and 1740 Bq/m^3), respectively, while the travel times were 1555, 1260, and 174 years, respectively. The large decrease in travel time between sensitivity cases INF02 and INF03 was not matched with a similar increase in concentration; however, the difference in infiltration rates between INF02 with 5.2 mm/yr and INF03 with 100 mm/yr is large and the largest maximum concentration at the downgradient point of calculation may be found with an additional sensitivity case between the infiltration rates of INF02 and INF03. Bounding the uncertainty of potential changes to the climate or land use within 1000 years, and within 10,000 years, would need to be part of a scenario development. The uncertainty in future infiltration rates and subsequently recharge rates would not only have an effect on the travel time of radionuclides within the unsaturated zone but on numerous FEPs including changes that may drive contaminant concentrations down (e.g., greater aquifer thickness).

If current climatic and hydrogeologic conditions do not significantly change, all alternative models performed such that the results did not appreciably diverge from the results of the base case model. Although in RPP-CALC-60793 (2016), DOE documented simulations that projected impacts of past waste releases that showed a peak concentration less than the Case 1a (a case with similar model and boundary settings as the base case except with water table level closer to the present than the assumed future). It also showed an earlier arrival of ^{99}Tc at the water table than had been observed in monitoring wells. Alternative Geologic Model II was judged by DOE to be more realistic than the heterogeneous alternative model.

Although DOE did concentrate much effort on demonstrating that the WMA C PA can adequately simulate transport in unsaturated zone, most of this effort was directed at the Hanford H2 sand unit. As previously discussed in Section 3.4.1.2.1 of this report, DOE responded (DOE, 2019) to NRC's RAI 2-9 (NRC, 2019) when DOE modified the original sensitivity case GRT4 of the concrete shell and infill grout properties from Hanford H2 sand values to hydraulic property values similar to those of the surrounding gravel-dominated backfill material while holding the infiltration rate constant. DOE described the adjustments that were made in order to obtain the results of the modified sensitivity case GRT4 (RPP-CALC-63407, 2019). DOE showed the STOMP model-generated results 1,000 years after closure in terms of the Darcy flux vectors as seen Figure 3-22 below. Figure 7-19 in RPP-CALC-63407 (2019) presents the volumetric aqueous moisture content around the representative tanks.

In Section 6.3.2.4 of the WMA C PA DOE had stated that each source term is transported to the saturated zone assuming that vertical mass transport in the unsaturated zone stays within the footprint of the source area, ignoring any lateral dispersion. Although lateral dispersion may not cause lateral movement as seen in Figure 3-22, this sensitivity case does not remove the difference in hydraulic properties between the gravel-dominated backfill material, the Hanford H1 gravel unit, and the Hanford H2 sand unit. Lateral Darcy flux vectors are visible in the Figure 3-22 cross-sections, especially so in the tank row consisting of tanks C-110, C-111, and C-112. Although contaminants moving in the H2 sand unit would appear to be gravity driven and thus be able to stay within the footprint of the source area, contaminants exiting from tank C-110 would not be staying within the tank's footprint while being transported in the gravelly backfill and H1 gravel unit. Lateral transport appears to be decreasing the footprint of the C-110 source area while increasing it for C-111. Since the figures provided by DOE are only for two tank rows and are two-dimensional, the magnitude of radionuclide mixing above the H2 sand unit is not clear (RPP-CALC-63407, 2019). The significance of potential lateral movement and transport above the Hanford H2 sand unit does not seem to have been investigated by DOE and the potential effects of such a process do not appear to be known. Since DOE assumes that each source term is vertically transported in the vadose zone within the footprint of the source area and ignoring any lateral dispersion (DOE, 2016), potential lateral movement above the H2 sand unit should be investigated and contaminant transport quantified. Therefore, NRC staff recommends that for future WIR evaluations and assessments, potential lateral movement and transport above the H2 sand unit be investigated, and if model results show this process to be occurring, that the significance of this process should be quantified (Recommendation #33).

Hydraulic property values that influence the pore-water velocity in the unsaturated zone were analyzed and assessed based on results obtained from the alternative conceptual models for the unsaturated zone. NRC staff concludes that due to these results and other similar analyses, base case values associated with the unsaturated zone have adequate technical bases. However, NRC staff also concluded that the analyses for sensitivity and importance of barriers would have provided more confidence if they had encompassed a broader range of values to bound uncertainty. Uncertainties in unsaturated zone hydraulic properties are derived from laboratory measured soil-moisture retention and unsaturated hydraulic conductivity datasets. These were fit using van Genuchten-Mualem constitutive relationships to derive various unsaturated zone hydraulic property uncertainty ranges and geometric mean values. Because WMA C-specific hydraulic properties data were unavailable, the above-mentioned hydraulic properties datasets for the Hanford H2 sand and for the undifferentiated H3/CCU/RF unit were populated with data originating from areas outside of the WMA C. Information from the three datasets (i.e., for the H2, the H3 and H1, and the backfill units) was then used to fit the derived saturated hydraulic conductivity values to a log-normal distribution allowing an estimate of a mean and standard deviation. The distribution was truncated at the minimum and maximum values of the data. A similar approach was used to obtain values for the van Genuchten parameters and the saturated and residual moisture contents for all three units. DOE is effectively assuming that the surrogate data is completely representative with no uncertainty by truncating the distributions at the minimum and maximum values of the observed data. Therefore, considering that the information obtained for each of the datasets used to derive uncertainty in the unsaturated hydraulic parameters was not from WMA C, NRC staff recommends that for future WIR evaluations and assessments, DOE not truncate the probability distributions at the minimum and maximum values of the observed data and use a broader range of values to bound uncertainty for unsaturated hydraulic parameters (Recommendation #34). A broader range would be more appropriate since it is uncertain if values less than the minimum or greater than the maximum would have been obtained if more samples had been gathered, or if samples had been taken from the actual site. On the other hand, confidence is provided by comparing the minimum and maximum values with the values in Table 6-14 from the WMA C PA (2016). This table shows the van Genuchten-Mualem parameter values associated with the maximum pore-water velocity based on the cumulative distribution functions in Section 8.1.4 of the WMA C PA. Most of these values lie between the maximum – minimum range. The exception is the saturated moisture content value from Table 6-14. This value is equal to the minimum value for all three geological units (H2, H1 and H3, and backfill), signifying that a greater range would have been more appropriate.

As discussed in the beginning of this section, the H2 Hanford sand distribution coefficient, or Kd, for ^{99}Tc had a correlation close to one and was the most important barrier that affected peak dose within a 1000-year timeframe (see Table 8-13). These results are based on an uncertainty range of Kd values from 0.0 to 0.1 ml/g. The base case Kd value in the PA is equal to 0 ml/g, which means that the ^{99}Tc radionuclide is moving as fast as the water is flowing. While NRC staff agrees that the conservative Kd value chosen for ^{99}Tc is appropriate, as are most of the Kd values and ranges used in the PA, there are some Kd values that may not adequately cover the range of uncertainty, in particular Se (selenium), I (iodine), Ra (radium), and Pu (plutonium).

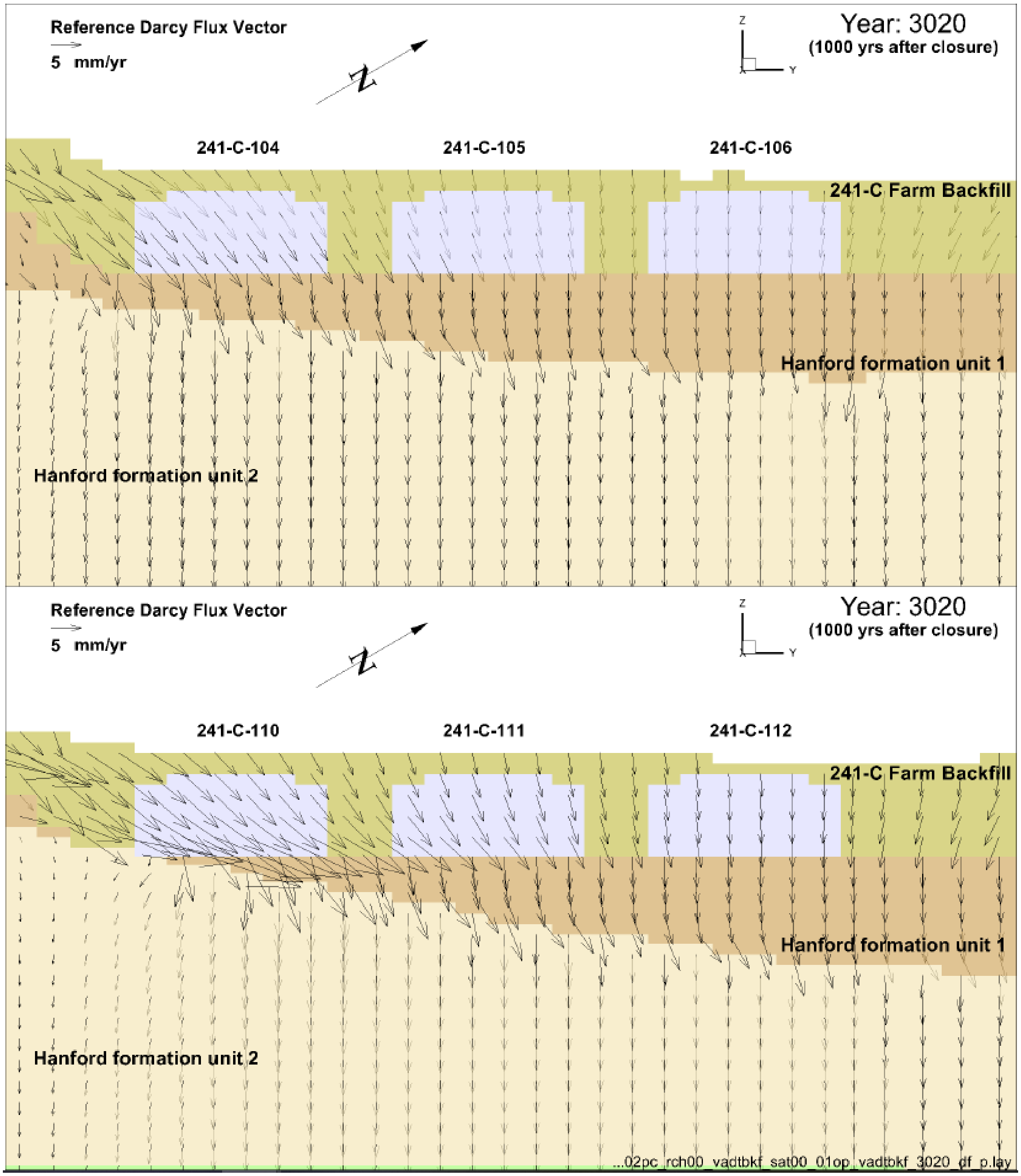


Figure 3-22 Visual Representation of the Flow Paths and Moisture Distribution in and Around the Tanks from the Modified Sensitivity Case GRT4 Results
[RPP-CALC-63407 (2019)]

In Table 4-9 of RPP-RPT-46088 DOE presented radionuclide distribution coefficient values based on measured values from 2003 and 2006 at the Hanford Site or estimated values from 2008 for WMA C (RPP-RPT-46088, 2016). The measured values for ^{79}Se at Hanford were between 0 and 1 ml/g with a best value of 0 ml/g and a recommended PA value of 0 ml/g; however, the value used in the WMA C model for the H2 Hanford sand is 0.8 ml/g. Tables 6-11 and 8-6 in the WMA C PA provided the references for the Kd values used in the PA including ^{79}Se at 0.8 ml/g. PNNL-17154 (2008) is given as the reference for selenium; however, this document is relying on sources from 1998 and expert opinion. Given the recommended Kd value of 0 ml/g from RPP-RPT-46088 (2016) and the deterministic nature of the STOMP model, a Kd value of 0 ml/g in the WMA C PA would be more appropriate.

The measured values from Table 4-9 in RPP-RPT-46088 (2016) for ^{129}I at Hanford were between 0 and 0.2 ml/g with a best and recommended value of 0.1 ml/g; however, the value used in the WMA C model for the H2 Hanford sand is 0.2 ml/g. PNNL-17154 (2008) is again given as the reference for iodine and this document in turn is relying on PNNL-14702 (2006) for the intermediate impact zone. However, PNNL-14702 (2006) states that, "For the intermediate impact zone, the best estimate is 0.1 ml/g."

The recommended value from Table 4-9 in RPP-RPT-46088 (2016) for ^{226}Ra from is 2 ml/g, and 3 ml/g is the Kd value taken from the Oak Ridge National Laboratory Risk Assessment System. Table 8-6 from the WMA C PA (2016) shows that the uncertainty analyses used a range from 5 to 10 ml/g with a base case value of 10 ml/g. The WMA C PA gives PNNL-17154 (2008) as the reference for radium which declared that no Ra studies for the unsaturated zone in the vicinity of a tank farm was available so that ^{90}Sr data from PNNL-14702 (2006) was used as a surrogate. In PNNL-14702 (2006), DOE stated that it was "expected that future work will incorporate ongoing multi-component ion exchange data to provide a more scientifically defensible approach for estimating Kd values for strontium-90." Based on the uncertainty associated with ^{226}Ra and its importance to peak dose after 10,000 years, a lower value of 2 or 3 ml/g recommended for ^{226}Ra from Table 4-9 (RPP-RPT-46088, 2016) should be included in the uncertainty range.

The measured values for plutonium at Hanford were between 62 and 620 ml/g with a "best" value of 190 ml/g and a recommended "denominator" of 5 (Table 4-9 in RPP-RPT-46088, 2016); however, in the WMA C PA uncertainty analyses, DOE used a range from 200 to 2000 ml/g with a base case value of 600 ml/g. The rate of plutonium transport in various environments is associated with significant uncertainty. In PNNL-21651 (2012), DOE described how recent research is pointing to the fact that acid transport may not explain the features of the plutonium and americium distributions in the Hanford Site 200 West Area (e.g. small concentrations of plutonium and americium have been found deeper in the unsaturated zone although these areas never received acid waste). Based on the uncertainty associated with the transport of plutonium and its general prevalence at the Hanford Site, the measured values for plutonium, with a minimum and maximum value of 62 and 620 ml/g, respectively, should be considered as the appropriate values to use in uncertainty analyses while the best value of 190 ml/g (Table 4-9 in RPP-RPT-46088, 2016) should be considered by DOE for the base case.

NRC staff concludes that some Kd values, such as ^{79}Se, ^{129}I, ^{226}Ra, and the plutonium isotopes, may not adequately cover the range of uncertainty given the information documented in RPP-RPT-46088 (2016). Therefore, NRC staff recommends that for future WIR evaluations and assessments, DOE should expand the uncertainty ranges for these isotopes (Recommendation #35).

Summary of Review

- Hydraulic property values that influence the pore-water velocity in the unsaturated zone were analyzed and assessed based on results obtained from alternative conceptual models for the unsaturated zone and the uncertainty analyses. If current climatic and hydrogeologic conditions do not significantly change, alternative models performed such that the results did not appreciably diverge from the results of the base case model used in the WMA C PA. NRC staff concludes that due to these results and other similar analyses, base case values associated with the unsaturated Hanford H2 sand unit have adequate technical bases.

- Based on the results obtained from the alternative conceptual models for the unsaturated zone and the associated uncertainty analyses, NRC staff has determined that the performance of the unsaturated zone safety functions such as thickness, sorption, and dispersion have adequate technical bases and support.

- Pore-water velocity in the unsaturated zone is an important barrier for delaying the arrival time of peak concentrations. Barrier performance of this feature would decrease with increasing precipitation rates.

- Recommendations #33 through #35 are discussed this section and included in Table 5-1 of this report.

3.7 Flow and Transport in the Saturated Zone

Flow and transport in the saturated zone are very important at the Hanford Site because the saturated zone (i.e., the groundwater aquifer) can significantly dilute the concentration of contaminants as the radionuclides travel to the offsite receptor.

3.7.1 Summary of DOE Analyses of Flow and Transport in the Saturated Zone

In the area of WMA C, the hydrogeological system basically consists of one aquifer. Although the upper portion of the basalt does contain water, most of groundwater within the basalt has an upward gradient and was not included in DOE's modeling effort. The unconfined aquifer system is in the undifferentiated H3/CCU/RF unit that overlies the basalt bedrock. The saturated thickness of the unconfined aquifer on the Hanford Site ranges approximately from 9 m to 12 m (30 ft to 40 ft) under the C-Tank Farms. The natural direction of flow beneath the WMA C is toward the southeast. However, the flow had, in the past, been in a northern direction due to water mounding from artificial recharge during operations at the Hanford Site. While the gradient is currently predicted to remain very flat under the WMA C (circa 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 thousands of meters per day, so that groundwater flux can be relatively high.

The natural, hydrogeological system at WMA C is a significant barrier limiting the impact of releases from the C-Tank Farm and is, therefore, a major component of the WMA C PA. DOE used the WMA C flow and transport model to simulate groundwater flow and contaminant transport in the saturated zone below the C-Tank Farm.

DOE identified safety functions for WMA C where a safety function is a feature of the system that provides a specific function that is relevant to the performance of the facility (see Section 3.2 of this report) (RPP-ENV-58782, 2016). In Appendix H in the WMA C PA (see Table H-1), DOE identified safety functions associated with the saturated zone. The performance of these safety functions may be related to the function's ability to dilute the concentration of the radionuclides by mixing uncontaminated water with contaminated water or by the ability of the media of the saturated zone (i.e., the undifferentiated H3/CCU/RF unit) to cling to, or to sorb, contaminants and prevent their movement. This section and the following section will discuss and evaluate performance of the following safety functions listed in DOE's WMA C PA: SZ1 - Water flow in saturated zone (dilution); SZ2 - Sorption on saturated zone soils (sorption); and SZ3 - Dispersion in saturated zone (dilution).

3.7.1.1 GoldSim One-Dimensional Probabilistic Model

DOE augmented the deterministic analyses with probabilistic analyses that used an abstracted model of the groundwater system. STOMP flow fields were used as inputs to the probabilistic GoldSim-based model. The GoldSim system model relied on the flow-field related parameters extracted from the STOMP model to provide moisture content, saturation, and Darcy flux input values. Figure 3-19 shows representative hydrostratigraphic columns for the tanks at WMA C (see Figure 6-47 in the WMA C PA). DOE compared the vertical discretizations of the GoldSim system-level model and the STOMP process-level model.

GoldSim was the primary software package used by DOE to perform the uncertainty analysis as documented in Section 8.1 of the WMA C PA. For the saturated zone, DOE described how the one-dimensional GoldSim model is oriented along the primary flow direction and was used for comparisons with performance objectives under DOE Order 435.1 (RPP-RPT-58948, 2016). Mass flux from the unsaturated zone was calculated for each source term (i.e., each single-shelled tank, the C-301 catch tank, the 244-CR vault, and pipelines) and was transported vertically to the aquifer without lateral dispersion. Flow and transport were lateral along the length of the aquifer pathway to the point of calculation (POC) 100 m (328 ft) downgradient of the WMA C fenceline which can vary between roughly 144 m (472 ft) to 235 m (771 ft) depending on source of the contaminants. The volumetric flow rate through the aquifer was calculated using the hydraulic gradient under steady-state conditions, the saturated hydraulic conductivity used in STOMP, the saturated zone thickness varying between 9 and 12 m (30 and 40 ft), and the width of the aquifer pathway. The width of the aquifer was taken to be the width of the source area if the source area is a tank (i.e., 23 m (75 ft)) or over half of the assumed pipeline source area length along the flow path (i.e., 75 m (246 ft)). Tanks C-102, C-105, C-108, and C-111 fall along a single flow path or stream tube since the flow direction was aligned with the orientation of these tanks and contained some of the highest tank residual inventories resulting in the highest concentration. The stream tube was used as a representative flow field for the 100-series tanks. Since overlapping plumes from the other tank stream tubes was not taken into consideration, additional contaminant mass from lateral dispersion along adjacent flow paths was added to the representative stream tube by using an analytical solution within GoldSim that calculates the concentration away from the centerline of the plume (i.e., the Plume function).

DOE undertook a spatial-variability study to evaluate the adequacy of using a representative flow field for the 100-series tanks and for the 200-series tanks (RPP-RPT-58948, 2016). The resulting observations supported the use of one representative flow field for all 100-series tanks, and another for all 200-series tanks. In addition, for the saturated zone, the highest concentration POC obtained by STOMP was compared to the GoldSim-based abstraction model results at 100 m (328 ft) downgradient of the WMA C fenceline along the tank C-105 aquifer flow path or stream tube. The one-dimensional aquifer pathway does not allow lateral dispersion of the mass, and therefore, the results showed slightly higher concentrations following the peak. DOE described this as sufficiently accurate for the intended use in evaluating system performance and indicated that the abstraction model was appropriate (RPP-RPT-58948, 2016).

3.7.1.2 STOMP Three-Dimensional Flow and Transport Model

The STOMP three-dimensional flow and transport model was used by DOE to estimate the future concentration in groundwater of various radionuclides and non-radiological contaminants to assist in evaluating the potential long-term impact on groundwater from residual waste left in tanks and ancillary equipment at WMA C. DOE documented the development of the numerical flow and transport model and the base case analysis (RPP-RPT-58949, 2016). In addition, the screening process used to identify and narrow the list of contaminants of potential concern that require evaluation in the STOMP model was also described. DOE discussed the input parameters and model development necessary to review the base case and screening calculations performed with environmental model calculations (RPP-CALC-60448, 2016).

The STOMP three-dimensional flow and transport model domain was rectangular in area with one side parallel to the general groundwater flow direction [length 738 m (2,420 ft)] and the other perpendicular to simulated groundwater flow [length 795 m (2,610 ft)]. The unsaturated and saturated zones in the base case were represented in the model by layers that were 104 m (341 ft) and 12 m (49 ft) thick, respectively. The unsaturated portion of the STOMP flow and transport model was composed of (from the top down) backfill material, H1 gravel, H2 sand, and the undifferentiated H3/CCU/RF gravels. Alternative Geologic Model II included an additional H2 gravel and H2 silty sand layer. The saturated zone was exclusively in the undifferentiated H3/CCU/RF unit. The vertical base elevation of the model was represented by the basalt and, consistent with the top of the model, the overall model thickness varied spatially according to the top of basalt elevation and surface relief.

A specified-flux boundary condition was applied at the top of the model and simulated recharge rates depend on site and surface conditions, the location and physical dimensions of WMA C, and the time of WMA C operations (Section 3.3 of this report discusses recharge). The bottom boundary of the unsaturated (vadose) zone was the water table and the bottom of the model (aquifer) was defined as a vertical no flow boundary condition (RPP-RPT-58949, 2016). Although the top of the relatively thin portion of the basalt is assumed to be a no-flow boundary, it has been established that there is a hydraulic connection between it and the Hanford formation, and that there are both upward and downward flows from and to the basalt-confined aquifer (PNL-10817, 1995). However, only a small amount of contamination has been detected in the upper basalt-confined aquifer and these are areas where confining units of basalt have been partially removed by erosion or are absent or where wells provided a pathway for migration. DOE assumed the basalt-confined aquifer system would not provide a pathway for contaminants from WMA C to the accessible environment (WMA C PA, 2016). No flow boundary conditions were also applied to unsaturated zone side boundaries and saturated zone boundaries running parallel to groundwater flow. Prescribed flux and prescribed heads exist in the aquifer on the upgradient and downgradient boundaries of the model, respectively. The prescribed flux was calculated based on the aquifer hydraulic conductivity and gradient, independent of recharge. The prescribed flux boundary condition value included a factor to account for the fact that the thickness of the unconfined aquifer varied because of the uneven surface of the underlying basalt. To account for the non-uniform aquifer thickness from the underlying basalt boundary and to keep the flux rate consistent throughout the groundwater model, the nominal flux rate was calculated by DOE as the product of the hydraulic conductivity and gradient (11,000 meters per day [m/d] times 2.0×10^{-5} m/m). DOE proportioned the flux according to the ratio of the average aquifer area perpendicular to the direction of groundwater flow throughout the model domain (9,440 m^2) and the aquifer area along the northwest boundary (6,151 m^2) where the prescribed flux was applied (i.e., the upgradient prescribed flux boundary was increased to 9,440 m^2/ 6,151 m^2 or 1.53 (see Table 3-12 for details).

Post-closure groundwater flow beneath WMA C is assumed to be northwest to southeast and parallel to the four tank arrays of 100-series tanks in WMA C. DOE stated that the justification for this assumption is found in RPP-RPT-46088 (2016). In RPP-RPT-46088 it was stated that interpretations of plume migration during the past several years would continue to support the general observation of closure or operational groundwater flow direction to the south and southwest beneath WMA C. Attempting to determine the pre-operational groundwater direction before the man-made recharge and contamination of the aquifer is difficult due to the small number of groundwater wells that were available to define water table conditions. Because of these limitations, the specific direction of groundwater flow in the immediate area of WMA C is difficult to determine and can only be inferred from the general regional interpretations of the water table provided prior to 1958. Based on the inferred regional water table interpretation, groundwater flow in the area south of WMA C was generally to the south and southeast. It is therefore assumed by DOE that once the water table decline of 0.11 m/yr (36 ft/yr) beneath WMA C has completed in 30 to 50 years, the groundwater direction will be the same. In addition, the topography of the top of the basalt, which forms the base of the unconfined aquifer in the area, can have up to 3.5 m (11 ft) of relief due to its erosional surface and influence local-scale flow directions especially in the shallower parts.

DOE's geochemistry conceptual model involved the partitioning behavior or sorption characteristics for release, retardation, and attenuation and simplifying assumptions for specific radionuclides (RPP-RPT-58949, 2016). The use of the constant soil adsorption coefficient, or adsorption-desorption distribution coefficient (Kd) model, was assumed by DOE to be generally applicable in the WMA C PA when contaminants are present at low concentrations. The Kd values were chosen assuming low-salt, near-neutral waste chemistry.

Table 3-12 Summary of Groundwater Domain and Characteristics Associated with Site-Specific Model Components for WMA C
[Table 4.4 in RPP-RPT-58949 (2016)]

Groundwater Domain and Characteristics	WMA C post-closure **water table elevation** ~119.5 m NAVD88 and average **hydraulic gradient** ~0.00002 m/m (Central Plateau Groundwater Model [CPGWM], CP-47631, 2015, "Model Package Report: Central Plateau Groundwater Model Version 6.3.3") Aquifer **hydraulic conductivity** = 11,000 m/day Aquifer area along **northwest cross-section** boundary = 6,151.04 m^2 (Model Grid Calculation) Aquifer area along **southeast cross-section** boundary = 13,997.55 m^2 (Model Grid Calculation) **Average** aquifer **area** along all aquifer cross-sections = 9,439.56 m^2 (Model Grid Calculation) Prescribed **flux along northwest** x-section boundary ($K_{sat.}$ = 11,000 m/d); 11,000 m/d × 0.00002 m/m × 365.25 d/yr = 80.36 m/yr; 80.36 m/yr × 9,439.56 m^2 / 6,151.04 m^2 = 123.31 m/yr Prescribed **head along southeast cross-section** boundary = 119.49 m (CPGWM, CP-47631, 2015 and Model Grid Calculation) Groundwater **thickness** is ~12 m (39 ft) in the aquifer (CPGWM, CP-47631, 2015 and Model Grid Calculation); Groundwater **concentrations evaluated for upper 5 m** (16 ft) of the aquifer (Section 3.1.8, WMA C PA) Aquifer hydraulic **conductivity** horizontal to vertical a**nisotropy 10:1** Aquifer **dispersivity** horizontal to vertical **anisotropy 10:1**

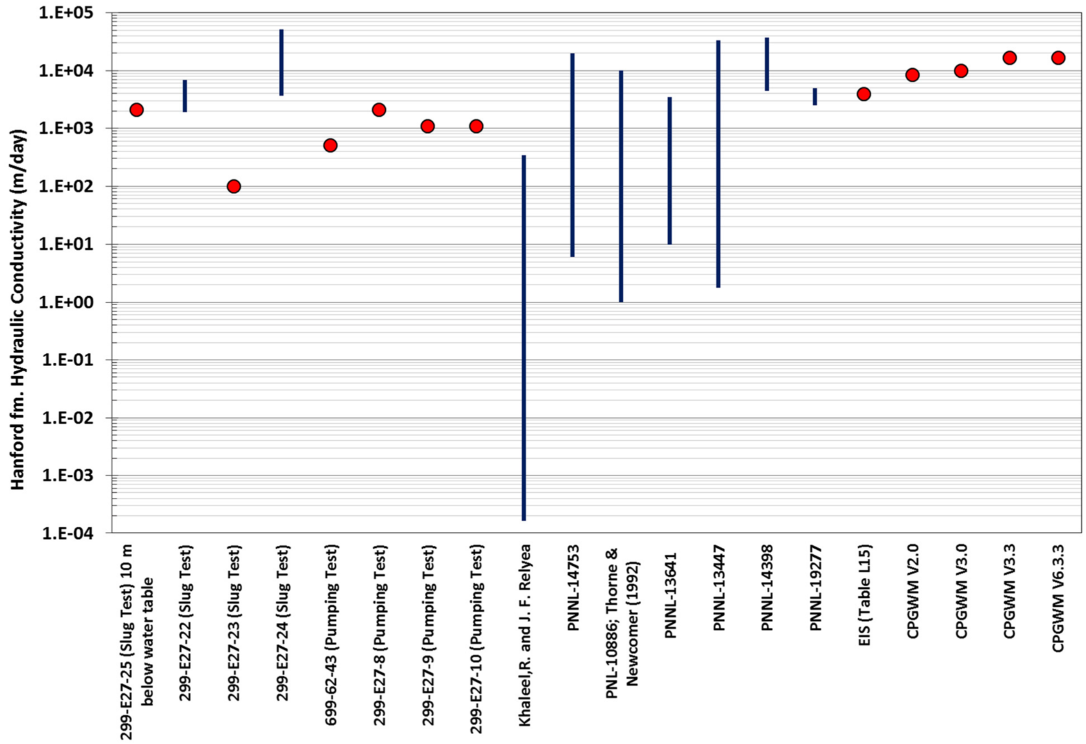
The geochemistry conceptual models for the Hanford Site are based on laboratory studies, testing, and measurements of adsorption and desorption coefficients under saturated and unsaturated conditions involving Hanford Site-specific sediments, contaminants, and conditions (DOE, 2016). DOE discussed how Kd values are typically lower for materials that contain significant amounts of gravel, such as the backfill and the units of H1 gravel and H3/CCU/RF (PNNL-17154, 2008). The basis for the Kd values used to approximate the transport of the contaminants and radionuclides, including gravel-corrected values, were presented in the WMA C PA (Table 6-11). The same Kd values were used for the saturated and unsaturated zone transport modeling.

A groundwater pathway screening analysis was used to reduce the number of radionuclides that needed to be simulated in the PA modeling by distinguishing those radionuclides that may impact groundwater during the specified compliance time period (i.e., 1,000 years) and the sensitivity and uncertainty analysis time period (i.e., 10,000 years). The screening process was used to determine the maximum Kd value of contaminants contained within the WMA C tank residuals that may reach the water table within 1,000 and 10,000 years. Those with non-gravel-corrected Kd values less than 3 mL/g were included because the results of the screening analysis indicated that radionuclides with Kd values greater than 3 mL/g did not impact groundwater within the 10,000-year sensitivity-uncertainty timeframe. A detailed discussion on the Kd values used in the WMA C and the appropriateness of these values is found in the preceding sections of this report on flow and transport in the unsaturated zone (Section 3.6).

Estimates of the Darcy flux or the specific discharge can provide valuable information about the hydraulic properties of the aquifer. In the WMA C PA, DOE indicated that few direct measurements of groundwater flux exist in the 200 East Area, "and none are particularly relevant to the groundwater flow conditions forecast for the unconfined aquifer in the immediate vicinity of WMA C." Since direct measurements of flux is difficult, DOE decided to obtain the flux by determining the values for hydraulic gradient and hydraulic conductivity (when multiplied together result in the groundwater flux). For the Hanford Site, the estimated hydraulic gradient is very small, with ranges between 1×10^{-5} m/m and 2×10^{-5} m/m, with current flow moving in a south to southeastern direction (WMA C PA, 2016). The average hydraulic gradient estimated from July 2011 through September 2012 was 2.5×10^{-5} ($\pm 0.4 \times 10^{-5}$) m/m toward the south and indicated that the hydraulic gradient determination represents a spatial average (SGW-54165, 2014). The recommended hydraulic gradient value from RPP-RPT-46088, Rev. 2 (2016) was 1×10^{-5} m/m.

Most of the flow and transport properties DOE used within the WMA C PA, including the hydraulic gradient, were obtained from results developed from the calibrated Central Plateau Groundwater Model, or the CPGWM, Revision 6.3.3 (CP-47631, 2015). One of the objectives for the CPGWM was to create a common modeling platform that can be used for investigations that support remedial activities and decisions. The CPGWM was not created as a single-time-use tool. DOE provided information pertaining to the CPGWM objectives; conceptualization; model implementation; sensitivity, calibration, and uncertainty analyses; configuration control; and limitations of the groundwater flow component of the CPGWM (CP-47631, 2015). Large-scale geologic and hydrogeologic features, including the extent of the paleochannel, were incorporated into the groundwater model to provide estimates of water levels, hydraulic gradients, and groundwater flows throughout the 200 West and 200 East Areas for current and expected future groundwater conditions. DOE used water level data relatively unaffected by anthropogenic activity from the 1940s, early 1950s, and first decade of the 21st century as calibration targets to estimate hydraulic properties. DOE compared the declining observed heads and CPGWM-simulated heads for a twenty-year period for two observation wells near WMA C in Figure C-11 of the WMA C PA.

The hydraulic gradient of 2×10^{-5} m/m used by DOE is close to the 2.5×10^{-5} m/m estimated in SGW-54165 (2014) but a factor of two higher than the value recommended in RPP-RPT-46088 (2010). It is based on the CPGWM estimates of future conditions (year 2200) within the Central Plateau. Water levels in the 200 East Area continue to decline, and evaluation of current flow direction and rate of groundwater flow at WMA C is difficult due to the very low hydraulic gradient. However, according to DOE, no appreciable change in hydraulic gradient is expected to occur after approximately 100 years after closure of WMA C.



References:

"Correcting Laboratory-Measured Moisture Retention Data for Gravels" (Khaleel and Relyea 1997).

CP-47631, "Model Package Report: Central Plateau Groundwater Model Version [as amended]."

DOE/EIS-0391, "Final Tank Closure and Waste Management Environmental Impact Statement for the Hanford Site, Richland, Washington."

PNL-10886, "Development of a Three-Dimensional Ground-Water Model of the Hanford Site Unconfined Aquifer System: FY 1995 Status Report."

PNNL-13447, "Transient Inverse Calibration of Hanford Site-Wide Groundwater Model to Hanford Operational Impact – 1943 to 1996."

PNNL-13641, "Uncertainty Analysis Framework – Hanford Site-Wide Groundwater Flow and Transport Model."

PNNL-14753, "Groundwater Data Package for Hanford Assessments."

PNNL-14398, "Transient Inverse Calibration of the Site-Wide Groundwater Flow Model (ACM-2): FY 2003 Progress Report."

PNNL-19277, "Conceptual Models for Migration of Key Groundwater Contaminants Through the Vadose Zone and Into the Unconfined Aquifer Below the B-Complex."

Thorne & Newcomer (1992) = PNL-8337, "Summary and Evaluation of Available Hydraulic Property Data for the Hanford Site Unconfined Aquifer System."

Figure 3-23 Estimates for Saturated Horizontal Hydraulic Conductivity Based on Slug Tests, Pump Tests, and Model Calibration
[Figure C-6 in the WMA C PA (2016)]

Therefore, DOE indicated a single value of hydraulic gradient for the water table was justified. Figure 3-23 (Figure C-6 in DOE, 2016) shows the various estimates and the range of estimates for saturated horizontal hydraulic conductivity based on slug tests, pump tests, and model calibration documented in numerous reports. The recommended hydraulic conductivity value from Table 5-11 in RPP-RPT-46088, Rev. 2 (2016) is 3000 m/d (9800 ft/d) for the unconfined aquifer for WMA C. In the WMA C PA, DOE relied on hydraulic conductivity values that were obtained from the calibrated CPGW Model (CP-47631, 2015) to calculate modified saturated hydraulic conductivities for the area around WMA C.

The domain of the CPGW model encompasses much of the Hanford Site and is well over 20 km (12 mi) long and 10 km (6 mi) wide. The WMA C is in the central eastern portion of the groundwater model as can be seen in Figure 3-24. The hydrogeologic units in the CPGW model near WMA C contain, from the top down, the Hanford formation, the Cold Creek unit, and the Ringold A formation, as shown in Figure 3-25 of this report. Differences between the hydrogeological units of the CPGW model and the STOMP model used in the PA include that the CPGW model had only one undifferentiated Hanford formation, while the unsaturated zone of the STOMP model is differentiated between the H1 gravel, H2 sand, and the H3/CCU/RF unit. The saturated zone of the CPGW model included the three hydrogeologic layers mentioned above, while the STOMP model only included the undifferentiated H3/CCU/RF unit. Figure 3-25 presents the extent of the three saturated hydrogeologic units within CPGW model layers in the vicinity of WMA C (black colored cell squares represent no saturated zone present). The calibrated hydraulic conductivities of CPGW model layers are used to calculate hydraulic conductivities for saturated zone layers of the STOMP model and a planar rectangular window (300 m × 200 m (1000 ft x 700 ft)) containing six model cells and encompassing most of WMA C flow domain was used to represent WMA C when making these calculations. The brown rectangular window in Figure 3-25 is the same brown rectangle as in Figure 3-26. The WMA C outline with single-shell tanks is visible in the background of Figure 3-26.

The hydrogeologic units (i.e., Hanford, Cold Creek, and Ringold) representing the saturated CPGW model layers 3 to 7 can be seen in Figure 3-25 (black colored cell squares represent no saturated zone present). Planar rectangular window (300 m × 200 m (1000 ft x 700 ft)) containing six model cells and encompassing most of WMA C is shown in Figure 3-25 (and Figure 3-26) and represented by the brown rectangle. For the WMA C PA modeling, the unconfined aquifer was treated as an EHM and WMA C saturated media hydraulic conductivity values were obtained using the field-scale calibrated regional CPGW model that accounted for appropriate local-scale boundary conditions, flow configuration, and history matching of well head data (DOE, 2016). Averaging, weighted according to the thickness of each CPGW model layer, was used by DOE to estimate the EHM effective saturated hydraulic conductivity and the hydraulic gradient. An equivalent saturated hydraulic conductivity was estimated for the undifferentiated H3/CCU/RF gravels of the STOMP model from the calibrated hydraulic conductivities of the Hanford formation with 17,000 m/d (56,000 ft/d), the Cold Creek unit with 400 m/d (1300 ft/d), and the Ringold A formation with 4.8 m/d (16 ft/d) from the CPGW model (CP-47631, 2015). The effective hydraulic conductivity in STOMP for the entire aquifer at WMA C, using an EHM approach, was estimated to be 11,000 m/day (36,000 ft/d) (see Table 3-13). To obtain this value, CPGW model layer 7 will be used as an example. As can be seen in model layer 7 from Figure 3-25, the brown rectangular window contains an equal number of model cells of the three hydrogeologic units found in the saturated zone of the CPGW model (i.e., 2 cells of the Hanford formation, 2 cells of the Ringold A formation, and 2 cells of the CCU). The average hydraulic conductivity value would be the average of the calibrated CPGW values of these six cells: (2 x 17,000) + (2 x 4.8) + (2 x 400) = 5802 m/d (19,040 ft/d).

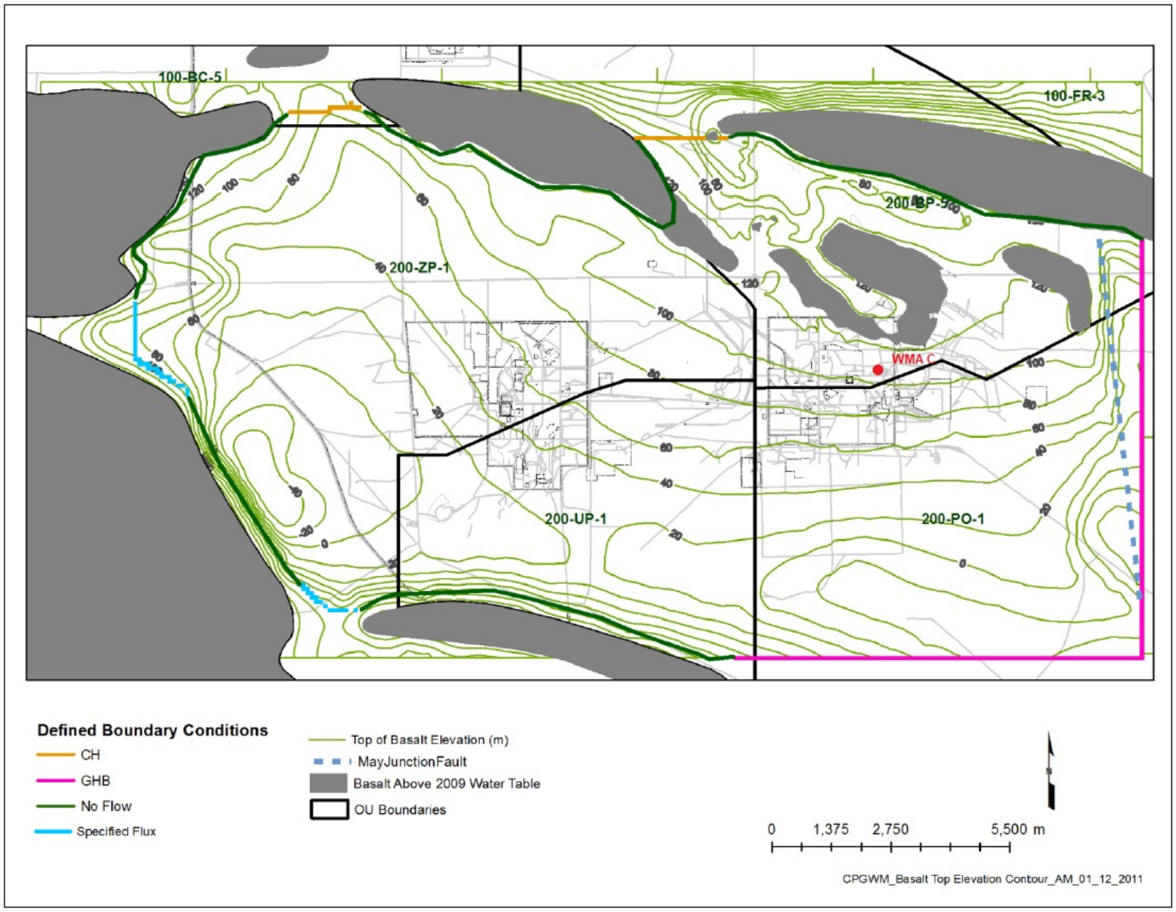


Figure 3-24 Boundary Conditions of the CPGWM
[Figure 4-3 in CP-47631 (2015)]

This value is representative of the bottom 1 meter of the 11-meter thick aquifer in the CPGW model. Adding the other values of the other 10 meters together with the bottom 1 meter gives a total of 123,102, which is then divided by 11 to obtain 11,191 m/d and rounded off to 11,000 m/d (36,000 ft/d).

Field-scale dispersivity was discussed by DOE in Section C.3.5.2 of the WMA C PA (2016). Gelhar et al. examined how heterogeneities at various length scales result in a scale dependence of dispersion (Gelhar, 1992). Figure 3-27 shows the dependence of field-scale dispersivity as a function of scale, with data classified by reliability. As can be seen in the figure, most data points with high reliability are 5 m (16 ft) or less. The WMA C PA examined three general relationships that quantify the dependence of dispersion on measurement scale. For the 100 m (328 ft) distance of transport from source areas to a compliance well located in the saturated zone, the calculated values of the three general relationships for saturated longitudinal dispersivity are equal to 17 m, 6 m, and 3.5 m. A value of 10.5 m (34.4 ft) was considered by DOE to be representative and the midpoint of the range of values. DOE selected a ratio of longitudinal to transverse field-scale dispersivity of 10.

Key assumptions made and limitations of the STOMP three-dimensional flow and transport model (DOE, 2016; RPP-RPT-46088, 2016; RPP-RPT-58949, 2016) and the CPGW model (CP-47631, 2015) as they relate to the saturated zone are:

- Post-closure groundwater flow beneath WMA C is assumed to be northwest to southeast and parallel to the four tank arrays of 100-series tanks in WMA C.

- Distribution coefficients (Kd) were used to represent sediment-contaminant chemical interaction that best represent plausible levels of reactivity. The Kd values were chosen assuming low-salt, near-neutral waste chemistry in the unsaturated and saturated zone.

- Applicability of the geochemical model was limited to radionuclides and non-radiological contaminants exhibiting linear isotherm behavior for contaminant release and attenuation. Surface complexation and precipitation was neglected.

- The POC used in the calculation of the groundwater concentrations corresponds to the location 100 m (328 ft) downgradient from the facility boundary per DOE Order 435.1.

- Results in the WMA C PA represent incremental groundwater contamination from WMA C residuals and do not include interaction with earlier WMA C waste releases.

- Applicability of WMA C model results are limited to evaluations where hydrogeologic parameter values remain constant and unchanging over time.

- For application of the CPGW model, the following limitations apply:
  - The flow model is regional in nature. Hydraulic property variation is generally recognized at the scale of kilometers to tens of kilometers horizontally. The eastern portion of the model is geologically more complex than the western portion of the model and the complexity of the former limits the scale for which simulated results should be considered reliable as evidenced by two observations:
    - Model calibrations indicate that there are some regions of kilometer scale, such as the northeast corner of the model domain, where flow is not well-represented.
    - Review of flow simulations in the 200 East Area, at less than a kilometer scale, have revealed very poor agreement with interpreted flow directions.
  - The model grid represents the aquifer with cells of dimension 100 by 100 m (328 ft by 328 ft). It is expected that the model is most suitable for making predictions of heads, hydraulic gradients, and groundwater flow rates over areas that comprise many model cells, and that predictions of these quantities on scales smaller than 100 m (328 m) are not reliable, except in circumstances of uniform hydraulic gradients.
  - Predictions made with the CPGW Model will be most reliable in those areas with a high density of water level data that were incorporated in the model calibration, and for those areas where model outputs correspond closely with the measured data. Conversely, model predictions will be less reliable in those areas where fewer water level data is available, as well as in those areas where model predictions do not closely correspond to measured data.

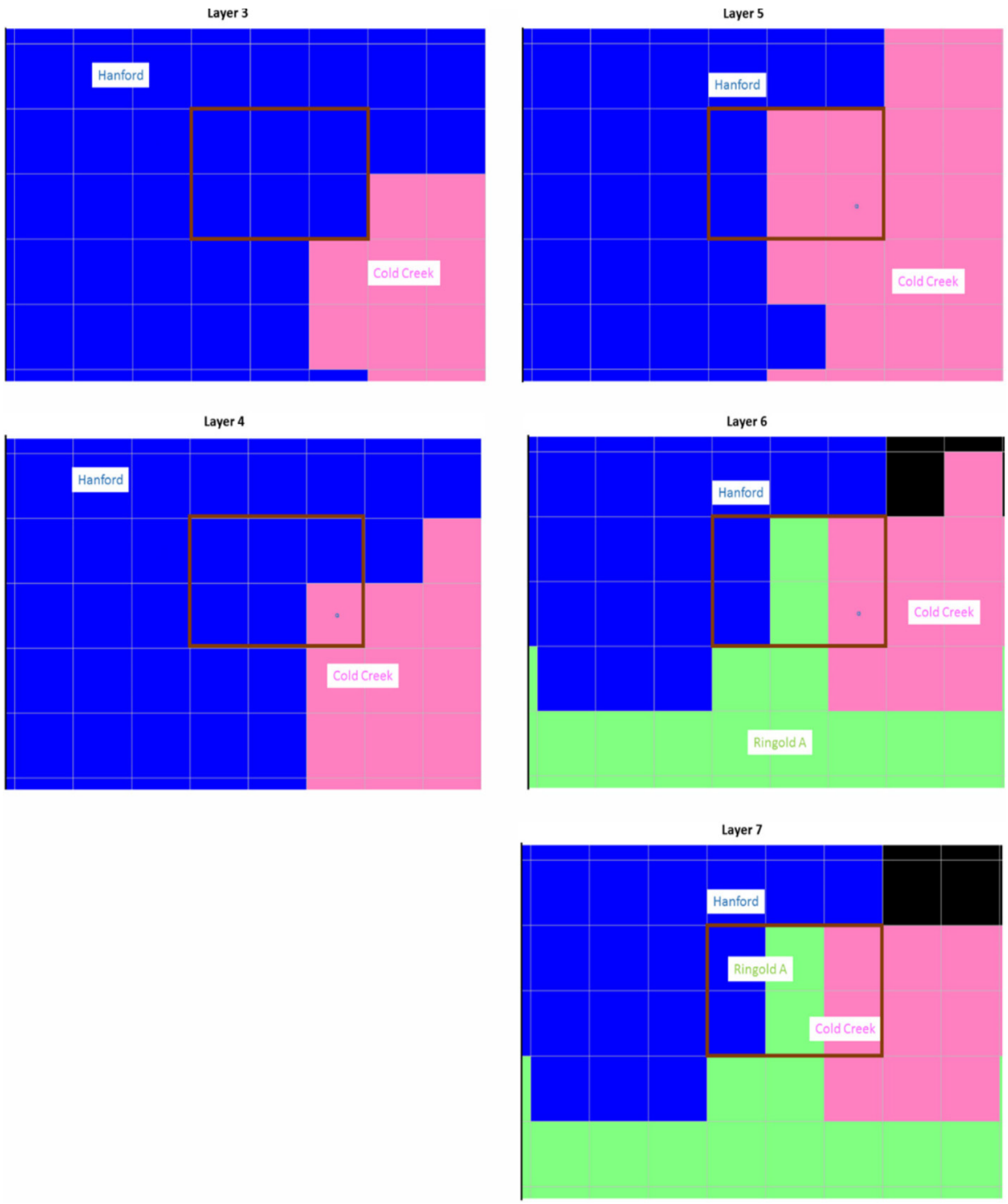


Figure 3-25 CPGW Model Layers 3 to 7 Showing the Hydrogeologic Units
[Figure C-9 in the DOE, 2016]

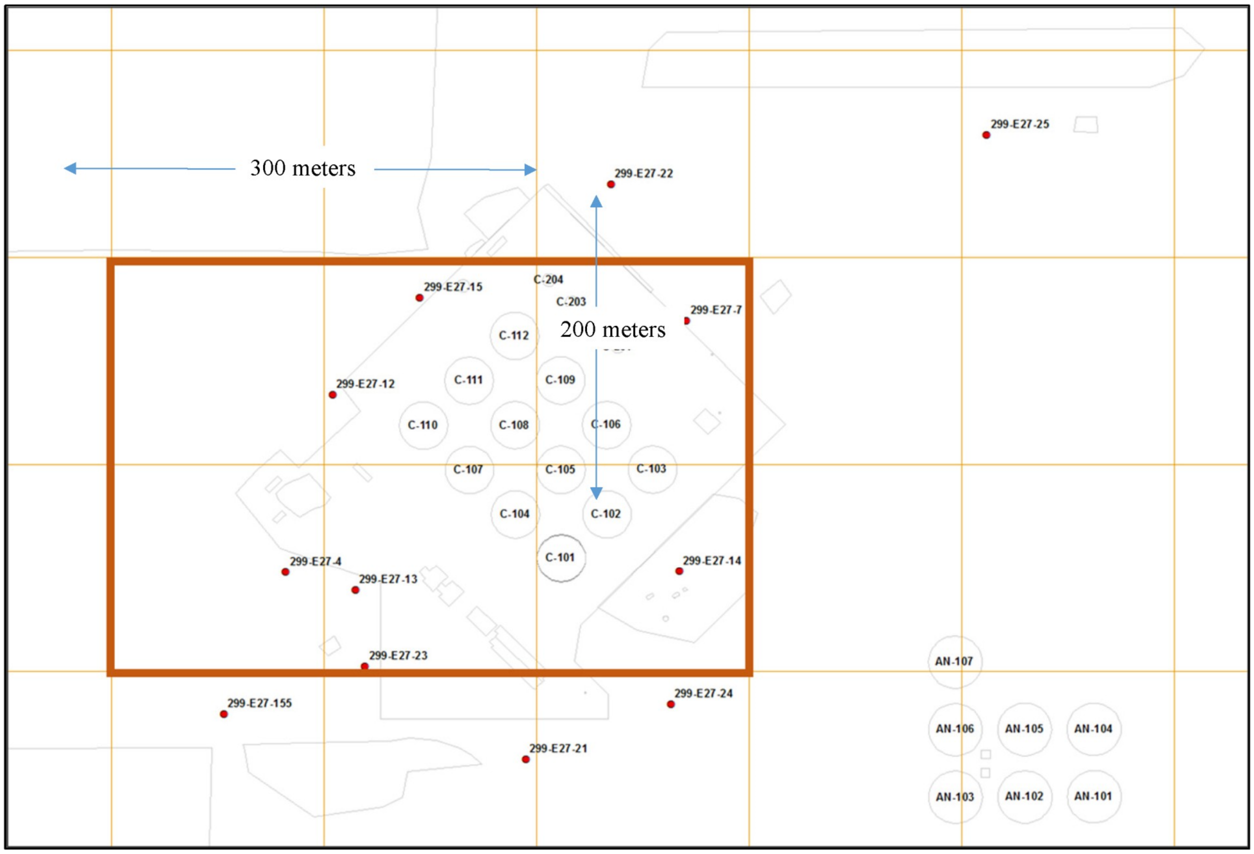


Figure 3-26 Planar Rectangular Window Containing Six Model Cells for Which the Weighted Average Hydraulic Conductivity was Calculated
[Figure C-5 in RPP-RPT-58949 (2016)]

In Appendix D of the WMA C PA DOE acknowledged that the assessment of uncertainties associated with how well models approximate actual relationships and conditions in the field is desirable, but that field data for model calibration is generally not available and/or attainable. Therefore, the WMA C STOMP flow and transport model was not calibrated by DOE. A comparison of measured and simulated water contents was carried out for the unsaturated zone in Appendix B, and parameters were adjusted accordingly. In addition, because PA models cannot be tested over the spatial scales of interest and the long time periods for which the models make predictions, the WMA C STOMP flow and transport model was not validated in the classical sense due to these long simulation times lasting hundreds and thousands of years.

In the WMA C PA, DOE stated that effective parameterization for WMA C saturated media hydraulic conductivity can be best achieved via a field-scale calibrated groundwater model, which accounted for appropriate local-scale boundary conditions, flow configuration, and history matching. For this reason, DOE estimates of hydraulic properties were based on the groundwater flux in the aquifer around WMA C, according to the calibrated CPGW model.

Table 3-13 Calculation of Layer Thickness Weighted Hydraulic Conductivity for the Planar Rectangular Calculation Window

[Table C-1 in RPP-RPT-58949 (2016)]

Year	Model Layer	Predicted Volumetric Water Flux (m^3/day)	Length of Window (m)	Layer Thickness (m)	Hanford Unit Calibrated Horizontal Hydraulic Conductivity (m/day)	Calculated Gradient (m/m)
2014	3	277.1	300	3	17,000	1.81×10^{-5}
2014	4	319.1	300	3	14,233	2.49×10^{-5}
2014	5	253.4	300	3	5,933	4.75×10^{-5}
2014	6	143.1	300	1	5,802	8.22×10^{-5}
2014	7	52.5	300	1	5,802	3.02×10^{-5}
2100	3	161.3	300	3	17,000	1.05×10^{-5}
2100	4	238.4	300	3	14,233	1.86×10^{-5}
2100	5	188.7	300	3	5,933	3.53×10^{-5}
2100	6	104.6	300	1	5,802	6.01×10^{-5}
2100	7	38.5	300	1	5,802	2.21×10^{-5}
Layer Thickness Weighted Hydraulic Conductivity (m/day, rounded):						11,000
Hydraulic Gradient 2014 (m/m, rounded):						3×10^{-5}
Hydraulic Gradient 2200 (m/m, rounded):						2×10^{-5}

The hydraulic property values derived from the calibrated CPGW model fluxes are modified and adapted and then applied to the STOMP flow and transport model domain. The CPGW model domain includes six hydrogeologic units with hydraulic property values established primarily through a transient calibration of the model to historical water level measurements. The CPGWM calibration process placed emphasis on matching water level data from the 1940s, early 1950s, and first decade of the 21st century to estimate hydraulic properties using flow conditions relatively unperturbed by site operations. Simulated water levels were compared to observed values for wells located upgradient and downgradient of WMA C. Matching the observed heads and the CPGWM-simulated heads over a time span of more than 20 years was intended to provide confidence in the predictive capabilities of the CPGWM and show how well the WMA C saturated media properties in the STOMP model are parameterized.

3.7.2 NRC Evaluation of Flow and Transport in the Saturated Zone

DOE performed a correlation analysis to identify uncertain parameters that influence the magnitude of peak dose regardless of the time. In this analysis, the saturated zone Darcy flux multiplier was the most important barrier with a correlation coefficient of -0.48.

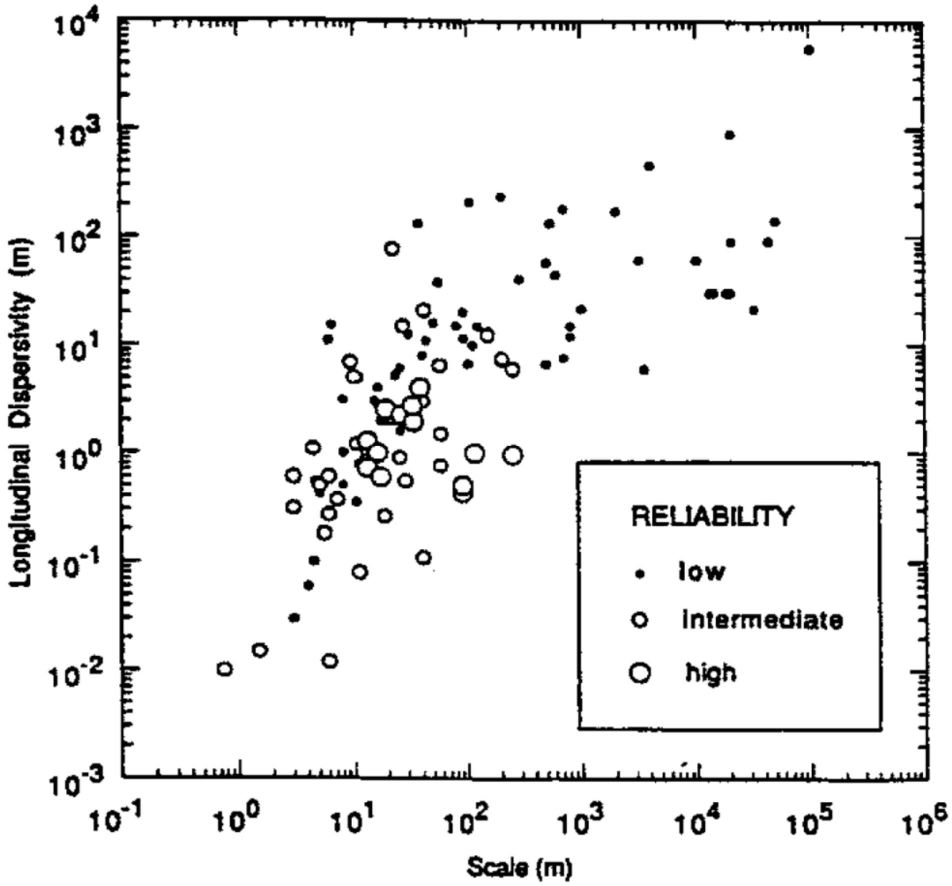


Figure 3-27 Field-scale Dispersivity in the Saturated Zone as a Function of Scale
[Gelhar et al., 1992]

Within 10,000 years, flow velocity ranked as the second highest parameter after pore-water velocity in the unsaturated zone. Within the 1000-year timeframe, dispersivity within the saturated zone ranked as the second highest parameter after the H2 sand formation distribution coefficient for ^{99}Tc. Multiple studies have shown that the certain parameters associated with the saturated zone can significantly affect concentration and dose results. Consequently, base case values associated with the saturated zone need to have technical bases, and analyses for sensitivity and importance of barriers need to encompass a sufficient range of uncertainty.

Most of the flow and transport properties used in the WMA C PA by DOE, including the hydraulic gradient, were obtained from results developed from the calibrated CPGWM (CP-47631, 2015). The planar rectangular window, as shown Figures 3-25 and 3-26, contains six model cells. This rectangular window was used by DOE to calculate the hydraulic conductivity value of the STOMP WMA C saturated model layer. The C-Tank Farm with single-shell tanks is visible in the background of Figure 3-25. Much of the WMA C is encompassed within the rectangular window as is an area to the east of WMA C. The area is equivalent to two CPGW model cells. However, this additional area to the east is not representative of the flowpath used by DOE to model contaminant concentrations. This flowpath is 100 m (328 ft) downgradient from the fenceline at the POC. An alternate approach that would encompass this flowpath would be to move the planar rectangular window containing the six model cells to the east to encompass this critical flowpath while still including WMA C (see Figure 3-26). Moving the window to the east would, however, change the representation of the hydrogeological units within that window (see Figure 3-25) based on the hydrogeologic framework of the CPGW model. The Cold Creek unit would be represented by a greater number of model cells while the Hanford units would be less represented. This would change the values used to obtain the effective hydraulic conductivity for the aquifer within the WMA C STOMP model. Using this averaging approach and following the same procedure are described in Section 3.7.1, the rounded calculated layer-thickness-weighted hydraulic conductivity obtained would be 5700 m/d (18,700 ft/d), roughly half the value shown in Table 3-13 (i.e., 11,000 m/d). As concentrations in the water well change approximately linear with the hydraulic conductivity in this model, this would result in approximately doubling the dose.

There are differences in the conceptual hydrogeological models of the CPGW model and the STOMP model, which makes a direct comparison between the Hanford Site model and the WMA C model difficult. A major difference in the conceptual hydrogeological models of the CPGW model and the STOMP model includes the three hydrogeologic layers representing the saturated zone in the CPGW model while the saturated zone of the STOMP model is represented only by the undifferentiated H3/CCU/RF unit. The CPGWM may represent the most recent culmination of understanding of the unconfined aquifer under the regional Central Plateau; however, the STOMP model represents the most recent culmination of understanding of the unconfined aquifer under WMA C. Parameter values associated with the geology and hydrogeology of WMA C are supported by field data, while the parameter values in the vicinity of the WMA C in the CPGWM are generally extrapolated due to the large scale of the CPGW model.

Although both the STOMP WMA C model and the CPGW regional model include the paleochannel as discussed in Section 1.2.3 of this report, the boundary of the channel within the CPGWM is different than that in the STOMP model due to the different model scales and differences in the grid block size between the two models. Theoretically, the paleochannel was a result of scouring, where the geological units such as the Cold Creek unit and Ringold formation, present below the WMA C, were eroded away due to the high velocity of the large flooding. The undifferentiated H3/CCU/RF is the byproduct of this scouring and represents the redeposition of the H3 Hanford gravel, the Cold Creek unit, and the Ringold formation into one undifferentiated unit. However, despite this difference and the fact that the thickness of the Hanford formation in Well 299-E27-14 is only 3 meters (10 ft) based on the stratigraphic framework of the CPGW model (see Table 3-12), the CPGW model was able to approximate the observed transient hydraulic heads for one WMA C upgradient well (Well 299-E27-15) and another WMA C downgradient well (Well 299-E27-14) for a period over 20 years (see Figure 3-4 in RPP-RPT-58949 (2016)). Although it is easier to simulate hydraulic heads on a flat gradient versus a steeper gradient, the differences in the transient match between the CPGWM simulated heads and the observed heads never appeared to more than a few centimeters or inches based on the diagrams in Figure C-11 in the WMA C PA (2016). Below the 3-meter thick Hanford unit, Well 299-E27-14 in the CPGWM is represented by the Cold Creek unit down to the basalt. The hydraulic conductivity value of the Cold Creek unit in the CPGWM is 400 m/d (1300 ft/d) while the Hanford formation has a value of 17,000 m/d (56,000 ft/d). It is not clear how the calibrated hydraulic conductivity values of the CPGW model would have changed if the STOMP stratigraphy had been used in the CPGW model near WMA C, but it does indicate that the set of parameter values and the variety of features used to obtain calibration for these two well heads are non-unique, which is not uncommon (i.e., a different hydrogeological framework and another set of hydraulic conductivity values could be used to obtain similar calibration performance).

Small-scale laboratory measurements provide the basis for hydraulic properties used to predict the large, field-scale flow behavior. This EHM approach replaces a heterogeneous geologic unit with a homogeneous one where upscaled or effective hydraulic properties are assigned. DOE applied an approach to the saturated zone that averages the calibrated hydraulic conductivities of the CPGW model Hanford formation and the CCU sediments. While the EHM approach used for the unsaturated zone accounts for the differences in scale between small, core-scale measurements and large, field-scale modeling, the approach used for the saturated zone is averaging different large, field-scale modeling values. The units being averaged to obtain a homogeneous value can be quite dissimilar, such as the Hanford formation with a calibrated value of 17,000 m/d (56,000 ft/d) being roughly 3,500 times more conductive than the Ringold formation with a hydraulic conductivity value of 4.8 m/d (16 ft/d). The groundwater regime, or flow net, created by interactions of these hydrogeologic features and flow processes is unique and would be changed if replaced with large-scaled features with different properties. Freeze and Cherry (1979) present a figure that illustrates this effect. Figure 3-28 shows the flowline refractions for two cases with two-layer units where the hydraulic conductivity is K1 = 10 x K2. The resulting flowlines traverse the low-permeability formation, or K2, by the shortest route and become almost vertical in any aquitard. High-permeability aquifers generally tend to have almost horizontal flowlines. In Figure 3-28, if contaminants were to hypothetically originate at the surface of the left box, the majority would exit from the relatively thin K1 layer on the right while for the right box the majority would exit on the right but nearer to the surface and in a less concentrated form. Averaging the two layers together would produce a third, incorrect, flow regime, and the location and scale of peak contaminant concentration would not be comparable to that produced by the original two cases.

Prescribed flux and prescribed heads exist in the aquifer on the upgradient and downgradient boundaries of the model, respectively. The upgradient, prescribed flux included a factor to account for the fact that the thickness of the unconfined aquifer varied because of the uneven surface of the underlying basalt. After the correction, the prescribed flux at the upgradient boundary was 53 percent higher than the flux at the middle of the WMA C model. In the response to NRC's RAI 2.13 (DOE, 2019), DOE maintained that the flux, or the flow velocity (hydraulic conductivity times hydraulic gradient), needs to be consistent throughout the groundwater model domain. The NRC staff questioned the need to maintain the flow velocity (length over time) but do agree that the groundwater flow, or volumetric flow rate (volume over time), that enters the model domain should equal the flow that exits the model domain as provided by the DOE and shown in Table 8 of ORP-63747 (DOE, 2019).

In Table 9 of ORP-63747, DOE showed that an average flux was not being maintained throughout the model domain since the SE boundary has a flux that is equal to about half that of the NW boundary (i.e., 0.15 m/d [0.49 ft/d] vs. 0.34 m/d [1.1 m/d]) although the average of these two boundary values is represented approximately in the middle of the model domain flowpath (ORP-63747, 2019). If the volumetric inflow rate and outflow rate needs to be maintained (i.e., what flows in must flow out) and the thickness of the saturated zone within the model varies, then the velocity at which the water travels must change to maintain mass balance (i.e., water velocity increases as available flow area decreases).

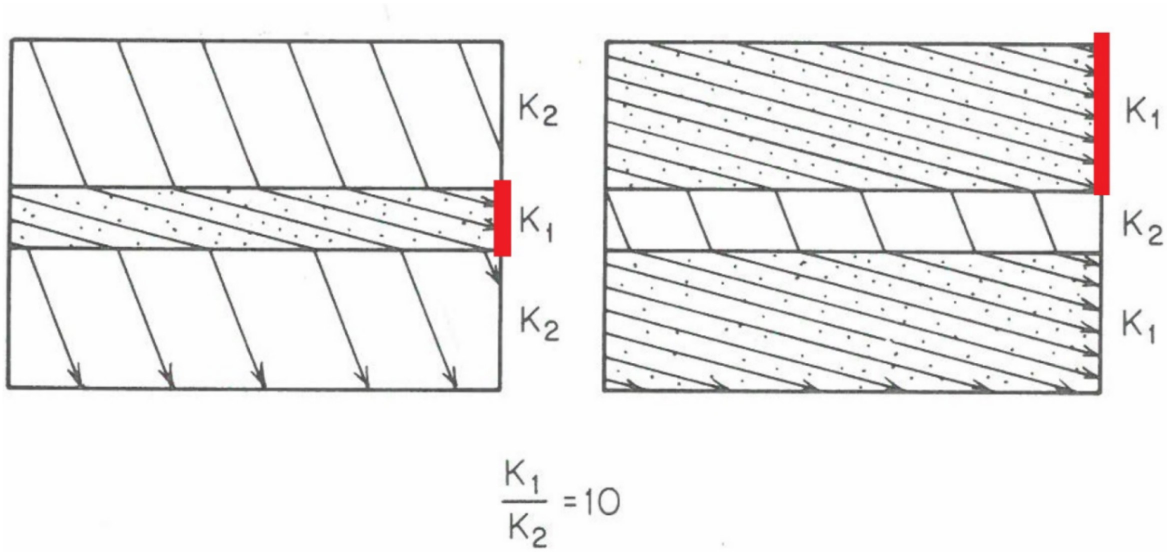


Figure 3-28 Refraction of Flowlines in Layered Systems with Differing Hydraulic Conductivities [Modified figure from Freeze and Cherry, 1979]

The NRC staff concludes that the technical basis provided by DOE for the hydraulic conductivity obtained by the layer-thickness weighted averaging approach was not sufficient. A technical basis is needed for using the relatively high hydraulic conductivity value for the unconfined aquifer obtained by the layer-thickness weighted averaging approach. The saturated hydraulic conductivity and the hydraulic gradient of an aquifer are important parameters for determining the degree of contaminant mixing and dilution in that aquifer, and the amount of dilution in the aquifer is a key safety function with respect to protection of offsite members of the public. In Section 8 of the WMA C PA, DOE showed that the saturated zone Darcy flux is an important parameter and the scale used can have a large influence on the final parameter value. DOE has stated that models calibrated with sufficient observations produce more accurate predictions than those based on smaller-scale estimated hydraulic conductivity values developed from hydraulic testing (DOE, 2019). DOE also states that hydraulic conductivities derived from calibrated models are regarded as more reliable for application to the WMA C PA model domain rather than direct measurements by permeameter, slug, or local-scale pump tests (less than 20 m) as such tests only investigate a very small portion of the aquifer that is not necessarily representative of the length scales evaluated in WMA C PA modeling.

The NRC staff is in general agreement that hydraulic property data obtained by direct permeameter, slug tests, or small-scale pump-tests measurements are not appropriate for model domains on the scale of kilometers or miles. However, the difference in scale between the CPGWM and the STOMP model is large. The model domain of the CPGW model encompasses much of the Hanford Site and approximately 2,000 square kilometers in area, while the scale of interest for the WMA C PA are the POCs 100 m (328 ft) downgradient from the WMA C fence line. Due to this large difference in scale, the NRC staff urges DOE to obtain corroborating data or additional model support to provide confidence that small-scale WMA C features or a localized zone of hydraulic conductivity that may influence the location and the level of concentration results are captured in the CPGWM. For example, in RPP-RPT-46088

(2016) DOE presented the results of slug tests (Table 5-3) showing low hydraulic conductivity values as well as high values within a single well (Well 299-E27-22) that vary by several orders of magnitude. Results for Well 299-E27-23 show a general low range of hydraulic conductivity values at 100 – 108 m/d (328 – 354 ft/d). On a regional scale similar to the CPGW model, features such as these would not play a significant role in the output; however, for results based on a transport distance of 100 m (328 ft), a zone of relatively low hydraulic conductivity may influence the results. Contaminants leaving the unsaturated zone and moving into the saturated zone at a location such as Well 29-E27-23 would be delayed in their transport before moving into the zones of the aquifer with higher hydraulic conductivities. Zones of higher hydraulic conductivity in Well 299-E27-22 were also relatively thin, usually less than a meter. This overall variability of hydraulic property values within and between the wells indicates that flowpaths within the aquifer may be tortuous and the mixing of waters within the total thickness of the aquifer would take longer than in an aquifer with homogenous hydraulic properties.

As shown in Figure 3-23, until recently, most DOE documents provided a range of hydraulic conductivity values that were lower than the current average WMA C PA value of 11,000 m/d (36,000 ft/d). Previously, in RPP-RPT-46088, Revision 1 (2010), DOE had given the general range of saturated hydraulic conductivity values for the unconfined aquifer as between 1,000 to 3,000 m/d (3,300 to 9,800 ft/d), with a recommended parameter value of 3,000 m/d (9,800 ft/d) and a recommended minimum and maximum of 100 m/d (330 ft/d) and 7,000 m/d (23,000 ft/d), respectively. The general range of hydraulic gradient values recommended for the unconfined aquifer was 1×10^{-5} m/m, with a recommended minimum and maximum of 2×10^{-6} and 1×10^{-4} m/m, respectively. This would provide an overall flow velocity or flux uncertainty range of 0.0002 m/d (0.0007 ft/d) to 0.7 m/d (2 ft/d). Even as recently as the "WMA C PA Mid-Year Status Working Session" in June 2014 (DOE, 2014), DOE listed the above hydraulic values as the hydraulic properties of the aquifer for the WMA C STOMP model.

In RPP-CALC-60793 (2016), DOE provided the results of WMA C flow and contaminant transport model simulations supporting scoping analysis and future projected impacts of past waste releases that indicate this larger range of hydraulic property values are justified. In RPP-CALC-60793 (2016), DOE stated that the intention of the analysis cases presented was to explore and evaluate the modeling assumptions and input parameter values that produce results consistent with the arrival times and general concentration levels of contaminants observed in monitoring wells around WMA C. Changes in the water table hydraulic gradient magnitude and flow direction during and after operations at Hanford lead to the sharp changes in the concentration data observed at the monitoring wells. Most breakthrough curve results from the analyses did not correspond to the rapid rise and fall of observed ^{99}Tc concentrations. Groundwater flux was one of the parameters tested in the analyses, as was the orientation of the gradient. That is, to address uncertainty in the gradient reversal from past Hanford operations, the release evaluation included both clockwise and counterclockwise rotation directions. As for the flow velocity in the saturated zone, the scoping analysis included two values of aquifer flux, the 10th percentile value (0.11 m/d or 0.36 ft/d) and the 90th percentile value (0.33 m/d or 1.1 ft/d). The evaluation of the 10th percentile aquifer flux and the transient aquifer conditions with a counterclockwise rotation of the hydraulic gradient provided the best approximation of the timing and magnitude of ^{99}Tc arrival in most of the monitoring wells surrounding WMA C. Assuming a hydraulic gradient equal to that of the base case, or 2×10^{-5} m/m, the hydraulic conductivity is then equal to 5,500 m/d (18,000 ft/d).

DOE decided to use the results of CPGWM (CP-47631, 2015) because it appeared to be the most appropriate and applicable calibrated model to provide values for parameterization, represented the most recent unconfined aquifer modeling efforts, and DOE had success in matching historical measured water level data from wells located upgradient and downgradient of WMA C for a time period spanning over 20 years. However, since output values from the CPGW model were used to populate some of the most significant STOMP parameters, documentation of the CPGW model development should have been integrated within the WMA C PA, similar to the development of the models for STOMP and GoldSim. CPGW model objectives, conceptualization, implementation, and application should have been documented in the PA as one of the primary models used. Limitations of the model results can be significant, and in CP-47631 (2015), DOE listed numerous limitations of the model that have a direct bearing on the use of the model results in the STOMP three-dimensional flow and transport model. These limitations involve the grid scale of the model and the eastern portion of the model is geologically more complex than the western portion of the model. Consequently, this can mean that flow is not well-represented in some regions of kilometer scale, such as the northeast corner of the model domain, which includes the WMA C. There is poor agreement with interpreted flow directions at less than a kilometer scale in the 200 East Area.

NRC staff recommends that for future WIR evaluations and assessments, documentation of the CPGW model development or a future analogous supporting model should be integrated within the WMA C PA including model objectives, conceptualization, implementation, and application. 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 (Recommendation #36).

The uncertainty discussed in this report can be evaluated with appropriate ranges of parameter values and alternative future scenarios and conceptual models. DOE performed uncertainty analyses such as the sensitivity analysis and barrier importance analysis (referred to in the WMA C PA, and therefore also in this report, as the uncertainty analysis) and documented the results in Chapter 8 of the WMA C PA. However, for some of the more potentially significant parameters, the values used in the analyses do not encompass the range that would capture the uncertainty associated with the saturated zone Darcy flux. For example, the hydraulic gradient and the saturated hydraulic conductivity were analyzed within the range of the groundwater flux (hydraulic conductivity times gradient) values. Increasing the flow velocity or flux increases dilution thereby decreasing contaminant concentrations. DOE had stated that the uncertainty in the Darcy flux multiplier is dominated by uncertainty in the saturated zone hydraulic conductivity. In addition, DOE stated that the hydraulic gradient is assumed to be constant after approximately 100 years at 2×10^{-5} m/m, based on the CPGWM estimates of future conditions within the Central Plateau. DOE indicated that the expected conditions justify the use of a single value of hydraulic gradient for the water table. However, DOE also stated in Section 8.1.3.6 of the WMA C PA that, “current monitoring has indicated that gradients can vary by a factor of two, due to Columbia River stage fluctuations and interconnections to the aquifer in the Central Plateau.” For hydraulic conductivity, in the uncertainty analysis DOE used a range of hydraulic conductivity values where 1,000 m/d (3300 ft/d) is the minimum value and 21,000 m/d (69,000 ft/d) the maximum value. The resulting flow velocity range is then from a minimum value of 0.02 m/d (1000 m/d time 0.00002), or 0.07 ft/d, to maximum value of 0.42 m/d (21,000 m/d time 0.00002), or 1.4 ft/d. This is in comparison to flux uncertainty range of 0.0002 m/d (0.0007 ft/d) to 0.7 m/d (2 ft/d) as presented at the “WMA C PA Mid-Year Status Working Session” in June 2014 (DOE, 2014); a minimum value 100 times smaller than the minimum flux value used in the uncertainty analysis. A stronger technical basis is needed to support DOE's PA base values, such as pump test representative of the approximately 100 m (328 ft) near the eastern edge of the central 200 East Area.

Field-scale dispersivity was rated relatively high in the results of the uncertainty analysis, as seen in Table 8-13 in the WMA C PA. The range in saturated zone dispersivity at the scale of the WMA C model is estimated by DOE to be from 1 m to 20 m (3.3 ft to 65.6 ft). However, as can be seen in the Figure 3-27, most data points with high reliability are 5 m (16 ft) or less, possibly indicating that 10.5 m (34.4 ft) is not a representative value of those values with high reliability. The range of dispersivity values used by DOE in the uncertainty analysis may be skewed towards the high end. The calculated values of the three general relationships for saturated longitudinal dispersivity are equal to 17 m, 6 m, and 3.5 m, or an average of 8.8 m or a median of 6 m. Since this parameter is ranked relatively high in importance at 1,000 years and 3,400 years after closure, the technical basis for the base case value of 10.5 m (34.4 ft) needs to be adequate or the range of values used in the uncertainty analysis needs to be sufficiently broad to account for the uncertainty. The base case value assigned by DOE lacks adequate technical and model support.

NRC staff recommends that for future WIR evaluations and assessments, DOE should provide a stronger technical basis for the saturated hydraulic conductivity value range. 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 (Recommendation #37).

NRC staff also recommends that for future WIR evaluations and assessments 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 (Recommendation #38).

Average concentrations within nine segments parallel to, and 100 m (328 ft) from the WMA C fenceline, are used to capture the maximum concentrations in the saturated zone. Concentrations calculated in the nine segments of the aquifer are assumed to be comparable to concentrations that would be measured by sampling a monitoring well at that location. The segments are approximately 30 m (98 ft) long and intended to align such that the plume centerlines for each row of single-shelled tanks parallel to the direction of groundwater flow is intersected by one of the segments. For example, Tanks C-103, C-106, C-109, C-112, and catch tank C-301 lie parallel to the general flow direction and their plume should combine to form one flowpath or stream tube. However, based on the figures shown in the WMA C PA, it is difficult to see if a particular flowpath aligns with an intended segment. Figures 7-22, 7-23, and 7-24 in the WMA C PA show the extent of the ^{99}Tc plume at different times. Distinct plumes with centerlines cannot be discerned. What is noticeable is that the flow and transport of the plume is neither completely in line with the three arrays of four 100-series tanks in WMA C nor perpendicular to the fenceline. This is confirmed by the data in Table 7-4 in the PA, which shows the maximum concentration of various contaminants at the POC for all the source structures. Contaminants from Tanks C-103, C-106, and C-109 are evaluated at POC segment 5 (width 27 m [89 ft]); however, the contaminants for C-112 and C-301 are evaluated at POC segment 6 (width 34.5 m [113 ft]). Tank C-111 is also evaluated in POC segment 5; however, the remaining tanks in that line, Tanks C-108, C-105, and C-102, have their maximum concentration in POC segment 4 (width 29.6 m [97.1 ft]). The plume centerline is not representative for all four tanks, but instead, for three tanks. Since the stream tubes do not align with the tank rows, and considering the importance of evaluating the maximum concentrations, the location and width of the segments should be analyzed for the potential to influence the concentration or dose results (i.e., sensitivity analysis). Therefore, NRC staff recommends that for future WIR evaluations and assessments, the location and width of the approximately 30 m (98 ft) long stream tube segments should be analyzed for their influence on the results (Recommendation #39).

As previously discussed, the thickness of the aquifer under WMA C may not remain constant for 10,000 years, but instead vary depending on climate, river water levels of the Columbia River, or human irrigation practices at or near the current WMA C. Saturated zone thickness should also be part of a sensitivity analysis in order to determine the significance of this parameter's capability to influence results. Therefore, NRC staff recommends that for future WIR evaluations and assessments, saturated zone thickness should be part of a sensitivity analysis in order to determine the significance of this parameter in influencing concentration or dose results (Recommendation #40).

Summary of Review

- The NRC staff finds that there is reasonable assurance that the performance objectives in 10 CFR Part 61, Subpart C, will be met. Alternative future scenarios with a lower water table or technical bases that support lower hydraulic conductivity values for the saturated zone units, and therefore, a reduced groundwater flux around WMA C, would substantially increase the peak concentration and dose; however, even if groundwater flux dilution was decreased by a factor of 110 (base case value of 11,000 m/d (36,000 ft/d) ÷ previous minimum recommended value of 100 m/d (330 ft/d)), peak concentration or dose objectives would be met based on given base case results.

- The key component of the average flux is the layer thickness weighted hydraulic conductivity value. The NRC staff concludes that a stronger technical basis is needed for using the relatively high hydraulic conductivity value for the unconfined aquifer. There are several sources of uncertainty associated with this value, which is the basis for the water available for dilution of contaminant concentrations [relevant to safety function SZ1 - Water flow in saturated zone (dilution)]. These uncertainties include: the placement of the CPGWM WMA C flux calculation window; averaging the different hydraulic conductivity values of the CPGW model layers representing H3, CCU, and Ringold A into one average layer value; and relaying on outdated hydrogeologic information in the CPGW model for the area surrounding the C-Tank Farm.

- The NRC staff questions the need to maintain the flow velocity (length over time) but agrees that the groundwater flow, or volumetric flow rate (volume over time), that enters the model domain should equal the flow that exits the model domain. There is no technical basis for increasing the original prescribed upgradient flow velocity at the northwest boundary in the STOMP model.

- Recommendations #36 through #40 are discussed this section and included in Table 5-1 of this report.

3.8 Biosphere Characteristics and Dose Assessment

The biosphere is the representation of the environment where humans may contact radioactivity that has been released from closed waste facilities. In the present-day, access 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 dose assessment takes the fluxes of contaminants to the environment and converts them into projected radiological impacts to receptors.

3.8.1 Summary of DOE Analyses of Biosphere Characteristics and Dose Assessment

DOE used point of assessment and timing assumptions consistent with the requirements in DOE Order 435.1 and the HFFACO. Site closure was assumed to occur in 2020, followed by a 100-year period of active institutional controls, which would prevent a member of the public from using the site and being exposed to radiation. Presently, DOE controls access to the site and no members of the public reside on the site. After the institutional control period of 100 years following closure, DOE assumed that the public could access the site. The point of assessment for the all-pathways dose analyses was 100 m downgradient from the fence line of WMA C. The concentration of radioactivity in groundwater was based on the model-simulated peak concentrations, accounting for aquifer dilution but not taking credit for wellbore dilution. The biosphere representation for the inadvertent intruder is discussed in Section 3.10 of this report.

The PA modeling combined the doses from the groundwater and air pathways and considered the receptor to be a hypothetical farmer. The farmer is a Representative Person (ICRP, 2006) who resides near the WMA C tank farm and uses contaminated water from a well. The water is used for consumption, to irrigate crops, and to provide water for livestock. The Representative Person is a person from the more highly exposed individuals in the population.

The Reference Person is a hypothetical aggregation of human (male and female) physical and physiological characteristics arrived at by international consensus for the purpose of standardizing radiation dose calculations (DOE, 2011a). DOE stated that the Reference Person replaced the average member of the critical group in the WMA C PA. The Reference Person is a concept used in dosimetry. The receptor is an adult who is assumed to receive exposures through the following exposure routes:

- Ingestion of water
- Ingestion of fruits and vegetables grown on the farm
- Ingestion of beef raised on the farm
- Ingestion of milk from cows raised on fodder grown on the farm
- Ingestion of eggs from poultry fed with fodder grown on the farm
- Incidental ingestion of contaminated soil
- Inhalation of contaminated soil (dust) in the air
- Inhalation of water vapor
- External exposure to radiation

DOE selected exposure parameter values and element-specific bioconcentration factors from a variety of references, documented in Sections 6.2.3 and 6.3.3 of the WMA C PA. Most of these parameters were generic and not based on site-specific measurements. Additional details were provided in RPP-ENV-58813, "Exposure Scenarios for Risk and Performance Assessments in Tank Farms at the Hanford Site, Washington." The dose conversion factors used were taken from DOE guidance documents (DOE, 2011a) and EPA's "Federal Guidance Report No.12, External Exposure to Radionuclides in Air, Water, and Soil" (EPA, 1993). Age- and gender-weighted intake rates were developed for a Representative Person. DOE used 95th percentile intake rates obtained from EPA/600/R-090/052F "Exposure Factors Handbook: 2011 Edition, National Center for Environmental Assessment" weighted by age and gender. Exceptions to this approach were the indoor inhalation rate and soil ingestion rates.

3.8.2 NRC Evaluation of Biosphere Characteristics and Dose Assessment

The NRC staff reviewed DOE's biosphere data, models, and documentation. DOE assumed the facilities will be closed in 2020, which is unlikely and, therefore, optimistic from a dose assessment standpoint. The time of facility closure has very little impact on groundwater dose impacts because short-lived isotopes decay during transport and do not reach the point of access. DOE included a 100-year period of active institutional control (to a date of 2120), which is consistent with NRC guidance and with industry practice in the commercial LLW disposal industry. Today, access to the Hanford Site is controlled by DOE, and environmental clean-up activities are likely to be continuing for a substantial portion of the 100-year institutional control period; however, it is uncertain how long DOE may control the site. DOE indicated that they planned to control the site in-perpetuity but did not rely on that assumption for their assessment. Doses from groundwater impacts are minimally affected by changes to the length of the institutional control period.

The receptor in the WMA C PA was defined as a farmer who resides in the area 100 m (328 ft) downgradient (or downwind) from the present day fenceline of WMA C. Figure 3-29 shows WMA C, the fenceline, and the point of dose calculations. The WMA sources are located variable distances from the fenceline, resulting in different transport lengths to the point of calculation. Tanks C-101 to C-103 are 43.86 m (144 ft) from the fenceline whereas Tank C-301 is 152 m (499 ft) from the fenceline. Though there are some differences in aquifer dilution and transport times from the release point to the point of calculation, the main impact of the different distances to the access point is on dispersion (which is dependent on length). In commercial LLW disposal, the NRC utilizes a buffer zone concept around waste disposal units. The buffer zone is the area of land that lies under the disposal units and between the disposal units and the boundary of the site. The buffer zone is assumed to be from 30 m to 100 m (98 ft to 328 ft). DOE's distances from the tanks to the point of calculation are longer than the buffer zone but are consistent with intent of NRC's commercial LLW disposal regulations. The Hanford Site is very large and has controlled access (as well as government land ownership). Land use at the assumed point of calculation at some time in the future is unknown but assumed to occur. This is likely a conservative assumption by DOE.

DOE took credit for dilution of contaminant fluxes that enter the aquifer from the vadose zone but did not take credit for wellbore dilution. When a well is pumped to withdraw water, a cone of depression or capture zone is formed around the well. This capture zone draws water in equal amounts from all directions resulting in some of the water being contaminated and some of the water being clean. DOE did not include these well dynamics in their evaluation, which is a conservative approach.

DOE's receptor was defined to be a farmer. NRC uses the average member of the critical group in assessments for 10 CFR Part 61 and 10 CFR Part 20. The critical group is determined (e.g., receptors located closest to the facility in a downgradient or downwind direction) and the average member has behavioral and metabolic characteristics consistent with the average characteristics of that group. DOE's use of a Reference Person is generally consistent with the use of an average member of the critical group. DOE used 95th percentile intake and consumption values. The water ingestion rate was 2.66 L/day (0.7 gal/day) and the crop ingestion rate was 272.33 kg/yr (600 lb/yr); use of these values would tend to result in a larger dose compared to the average member of the critical group with mean consumption values. However, DOE used a fraction of locally-produced crops (fruits and vegetables) that is consumed (0.25). If the critical receptor is a resident-gardener or resident-farmer, the fraction of produce that is locally-grown is generally higher and would have an upper bound of 1. The overall result of using 95th percentile intake values combined with lower locally-produced crops fraction is that DOE's implementation is generally consistent with the average member of the critical group, however, it is not a conservative approach for intake through crops.

With a depth to water of approximately 80 m (262 ft) and relatively arid conditions, subsistence farming in the vicinity of WMA C is less likely, albeit possible, compared to other more economical and environmentally practical locations. DOE land ownership, the presence of a large engineered closure cap (depending on the design), and the relatively sparse population density contribute to a decreased likelihood of a resident farmer receptor scenario. However, the uncertainty associated with long-term projections dictates a cautious but reasonable approach, such as the resident farmer scenario adopted by DOE. The exposure pathways that DOE included were consistent with NRC guidance and industry practice. Major pathways were included in the assessment. Other pathways are possible, however, are anticipated to only result in minor changes to the dose assessment results.

DOE should ensure that the models used to simulate the release and transport of radionuclides are consistent with the assumptions about the biosphere (Recommendation #41). For example, farming, especially commercial farming, may make use of widespread irrigation that would have the potential to significantly increase effective recharge rates and result in more rapid transport times through the vadose zone.

In 2009, the NRC staff issued RAIs after a review of an evaluation for Tank 241-C-106 (NRC, 2009). One technical issue addressed in that package was that DOE did not provide justification for excluding a receptor scenario for Native Americans. Native Americans have lived and continue to live in the vicinity of the Hanford Site and often have different habits, lifestyles, and consumption and exposure parameters. DOE responded that as part of the Comprehensive Environmental Response, Compensation, and Liability Act of 1980 (CERCLA) process for identifying exposed individuals, DOE made a request for Tribal Nation stakeholders to provide exposure scenarios that reflect their traditional activities (DOE, 2019).

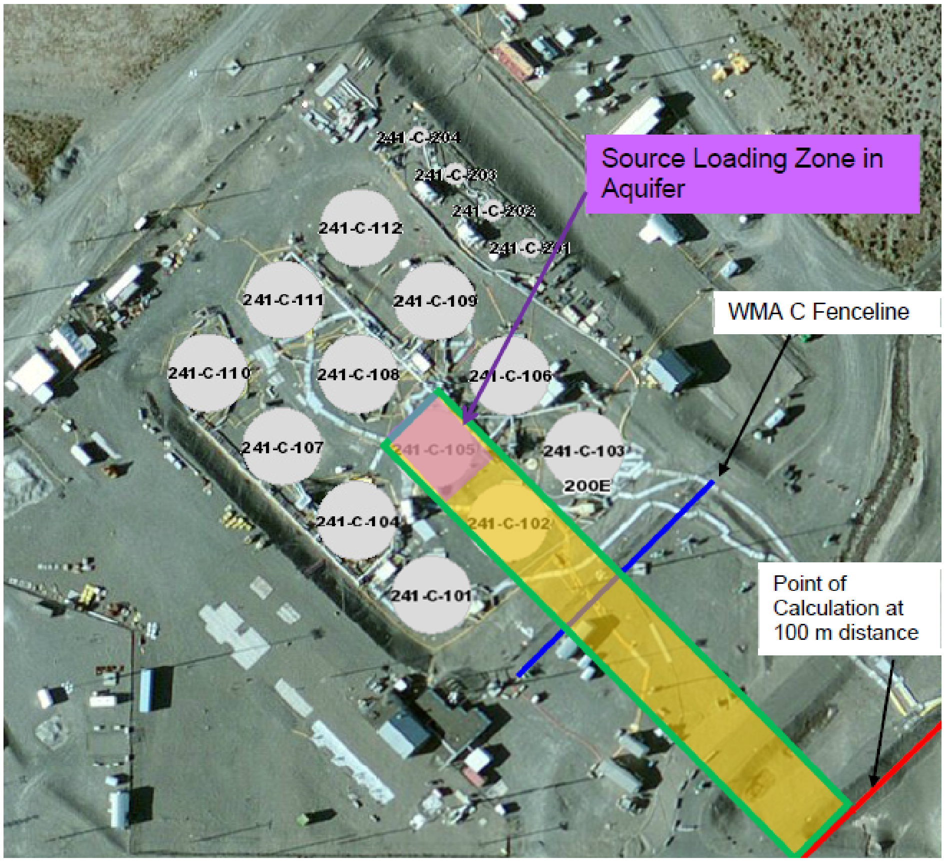


Figure 3-29 WMA C 100-series Tanks with Fenceline and Point of Calculation
(DOE, 2016)

DOE indicated that they respect the views of Tribal Nations and has considered them in the preparation of Remedial Investigation/Feasibility Study (RI/FS) and RCRA Facility Investigation/Corrective Measures Study (RFI/CMS) investigations at the Hanford Site. DOE agreed to include quantitative analyses of the Native American Assessment Scenarios in the RI/FS documents being prepared on the Hanford Site. The Native American exposure scenario impacts were provided in the document as additional risk management information to Tribal Nation stakeholders to assist them in providing potential mitigation measures at WMA C. Though this information was provided with respect to other decisions at WMA C, it was not provided for the Draft WIR Evaluation for WMA C. As noted above, the pathways and parameters for a Native American receptor can differ materially from a non-Native American receptor. For example, in HNF-SD-WM-TI-707, Rev 5 (2007) the combined consumption of locally-grown fruit and vegetables (excluding grain) is 169 kg/yr (372 lb/yr) for the Native American Subsistence Resident compared to 26.6 kg/yr (59 lb/yr) used for the onsite receptor (intruder). As previously mentioned, the crop ingestion rate for the offsite receptor (suburban gardener) was 272 kg/yr (600 lb/yr). It is recommended that DOE provide the dose results for Native American receptors at Hanford to increase transparency with potentially impacted stakeholder groups (Recommendation #42).

Table 3-14 Comparison of Select DOE DCFs with FGR-11 and ICRP-72 (mrem/pCi)

Isotope	DOE-Ing	FGR11-Ing	ICRP72-Ing	DOE-Inh	FGR11-Inh	ICRP72-Inh
^{241}Am	8.8×10^{-4}	3.6×10^{-3}	7.4×10^{-4}	1.6×10^{-1}	4.4×10^{-1}	3.6×10^{-1}
^{14}C	2.3×10^{-6}	2.1×10^{-6}	2.1×10^{-6}	8.2×10^{-6}	2.1×10^{-6}	2.5×10^{-5}
^{137}Cs	4.9×10^{-5}	5.0×10^{-5}	7.4×10^{-6}	1.7×10^{-5}	3.2×10^{-5}	1.4×10^{-4}
^{129}I	4.5×10^{-4}	2.8×10^{-4}	4.1×10^{-4}	1.5×10^{-4}	1.7×10^{-4}	1.3×10^{-4}
^{238}Pu	9.7×10^{-4}	3.2×10^{-3}	8.5×10^{-4}	1.7×10^{-1}	3.9×10^{-1}	4.1×10^{-1}
^{90}Sr	1.3×10^{-4}	1.5×10^{-4}	1.1×10^{-4}	1.5×10^{-4}	1.3×10^{-3}	5.9×10^{-4}
^{99}Tc	3.3×10^{-6}	1.5×10^{-6}	2.4×10^{-6}	1.6×10^{-5}	8.3×10^{-6}	4.8×10^{-5}
^{234}U	2.2×10^{-4}	2.8×10^{-4}	1.8×10^{-4}	1.4×10^{-2}	1.3×10^{-1}	3.5×10^{-2}

To convert mrem/pCi to mSv/Bq multiply by 0.2703. Inh = inhalation; Ing = ingestion.

Table 3-14 provides the DCFs for select radionuclides for ingestion and inhalation that DOE used compared to the values from FGR-11 and ICRP-72 (EPA, 1988), (ICRP, 1996). Newer values (e.g., FGR-13) are also available but the comparison is similar. DOE's primary radionuclide of concern is ^{99}Tc and the primary exposure pathway for ^{99}Tc is consumption. Though the DCFs for some radionuclides are lower than the other reference values, the DCF for ingestion of ^{99}Tc is higher. DOE did not include any variability or uncertainty in DCFs in their assessment, which is consistent with industry practice. However, it is important to understand that DCFs may be revised by EPA in the future, therefore, there is some uncertainty inherent in the dose calculation.

Summary of Review

- The NRC staff reviewed DOE's biosphere data, models, and documentation. DOE's consideration of biosphere characteristics and implementation in a dose assessment were appropriate to assess the potential impacts from waste residuals remaining in WMA C.

- There are no major sources of uncertainty associated with this section.

- Recommendations #41 and #42 discussed in this section are also included in Table 5-1 of this report.

3.9 Performance Assessment Results and Protection of the Public (10 CFR Part 61.41)

The requirements for protection of offsite members of the public from releases of radiation is provided in 10 CFR Part 61.41. A PA or other calculation is typically used to project potential doses to members of the public. This section summarizes the results of DOE's PA and the NRC staff's review of the information. The previous sections discussed the features, events, and processes that comprise the performance assessment calculations. This section provides the results of the calculations and NRC's review of those results with respect to the criteria.

3.9.1 Summary of DOE Performance Assessment Results

As discussed previously, DOE developed a PA model and supporting documents[9] to estimate potential dose results for an offsite member of the public to demonstrate the requirements in DOE Order 435.1 will be met with reasonable expectation (DOE, 2016). As discussed in Section 1.5, Criterion B specifies that the residual waste will be managed to meet safety requirements comparable to the performance objectives set out in 10 CFR Part 61, Subpart C.

DOE's analyses of performance included a deterministic base case, probabilistic uncertainty analyses, sensitivity analyses, and calculations to explore the performance of safety functions. In response to questions asked by the NRC staff during the review process, DOE completed additional analyses (DOE, 2019). DOE used the combination of all these analyses to demonstrate compliance.

Figure 3-30 provides the plot of doses for the probabilistic analyses and the deterministic base case for the groundwater pathway. The peak groundwater pathway dose within the 1,000-year compliance period that DOE used was 4×10^{-6} mSv/yr (4×10^{-4} mrem/yr). For the sensitivity and uncertainty analysis period, which extended to 10,000 years, the peak groundwater dose was 1.0×10^{-3} mSv/yr (0.10 mrem/yr); this is well below the standard of 0.25 mSv/yr (25 mrem/yr). Most of the groundwater dose was from ^{99}Tc, with lessor contributions from ^{79}Se, ^{129}I, ^{126}Sn, and uranium isotopes and their progeny.

The groundwater doses were more uncertain when breakthrough occurred, and less uncertain (less than 2 orders of magnitude between the 5th and 95th percentile results) at longer times. DOE indicated that even the dose result for the highest realization from the probabilistic analysis was lower than the standard. Using the methodology described earlier in this report, DOE calculated the peak air pathway dose to an offsite member of the public of 4×10^{-5} mSv/yr (4×10^{-3} mrem/yr) during the compliance period, primarily from the release of ^{3}H. The all pathway dose results for the deterministic base case showing the contributions from different radionuclides are provided in Figure 3-31.

To examine safety functions, DOE performed sensitivity analyses whereby the function of a barrier or process was altered to evaluate the potential impact on performance. DOE stated the purpose was to evaluate the effects of changing a broad set of input assumptions. The primary modeling assumptions evaluated included: natural system heterogeneities, long- term engineered surface and subsurface barrier performance, and human actions. Depending on what was being evaluated, DOE either used the STOMP model or GoldSim. The cases evaluated included five for surface barrier flow, two for aquifer dilution, five for vadose zone flow and dispersion, two for inventory, four for grout flow, one for residual chemistry, and three for tank flow, for a total of 22 cases. DOE calculated relative changes in downgradient water concentrations at the POC compared to the base case.

[9] The primary reference for the PA analyses is RPP-ENV-58782 Rev. 0 (DOE, 2016). The analyses were based upon numerous supporting references, many of which are found in the reference section of this TER. The notation "WMA C PA" is used in this section to refer to the primary reference or model.

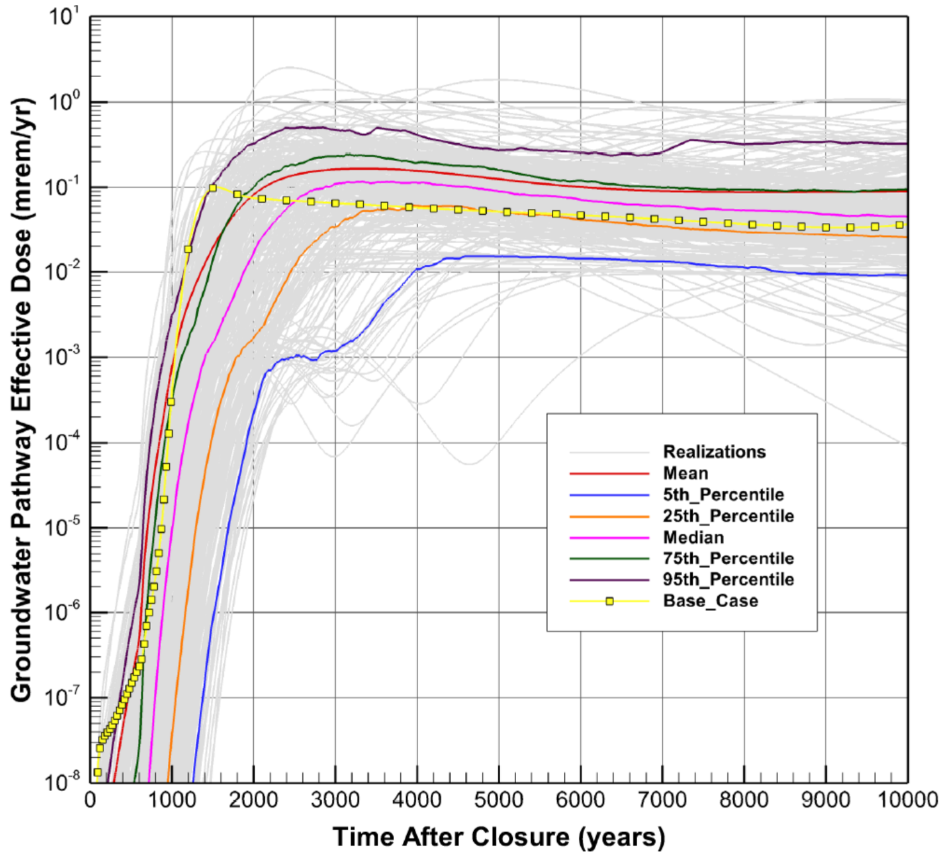


Figure 3-30 Groundwater Pathway Doses for the Probabilistic Analysis and Deterministic Base Case [Figure 8-23 in the WMA C PA]

Based on DOE's analyses, the effects of safety function changes included:

- Recharge had a small impact on concentrations of ^{99}Tc but could have a larger impact on the time of arrival when recharge rates were high.
- Inventory had, predictably, a directly proportional impact on groundwater concentrations (e.g., a factor of 5 increase in inventory increased groundwater concentrations by a factor of 5).
- Aquifer property changes (i.e., dilution) had a directly proportional impact on groundwater concentrations. The results ranged from 160% higher to 37% lower than the base case.
- Unsaturated zone parameter changes resulted in up to a 60% increase to a 13% decrease in groundwater concentrations compared to the base case.
- Grout flow analyses cases, which all looked at lessor performance, resulted in increases from 0% to 44% in groundwater concentrations, with the latter case being for degradation of the tank grout at 0 years after closure.
- Cases for changes to the release rate of ^{99}Tc had no impact.
- Tank flow safety function cases showed increases of up to 38% in groundwater concentrations for significant changes to grout diffusion coefficients.

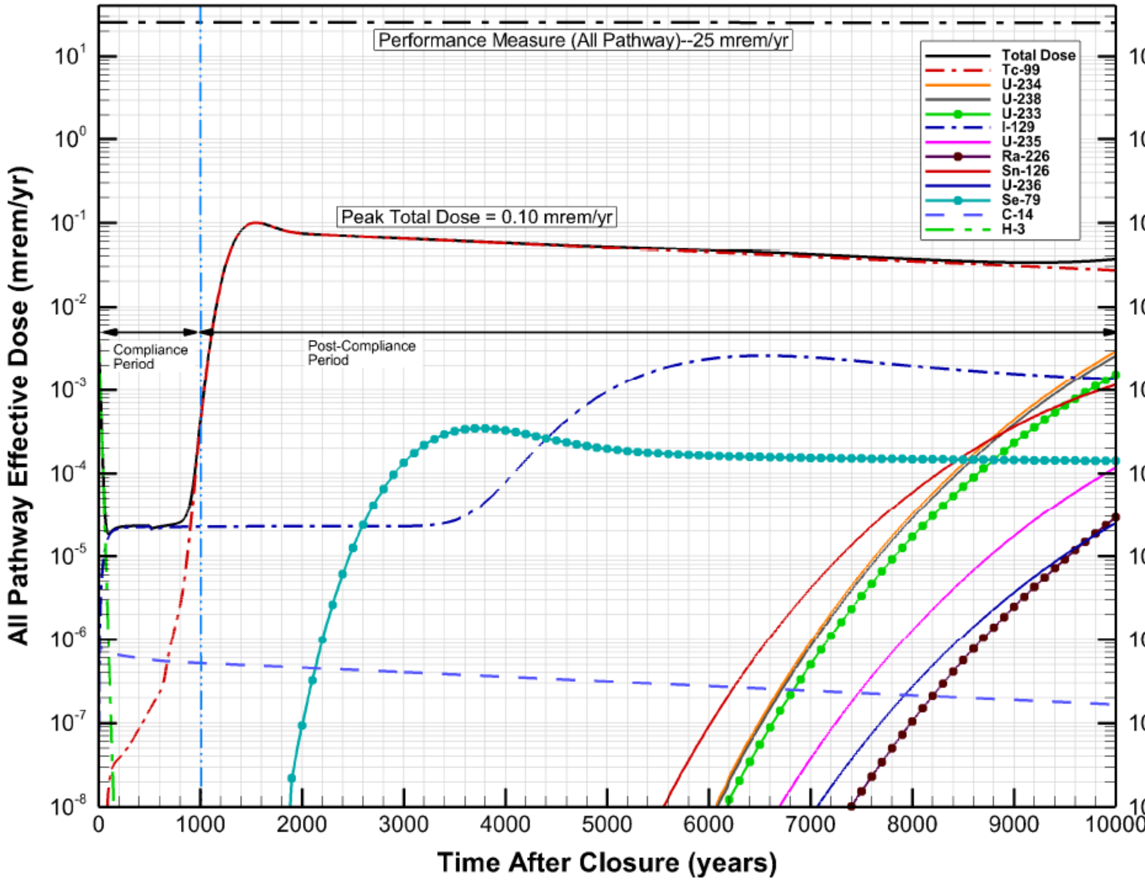


Figure 3-31 All Pathway Doses for the Deterministic Base Case
[Figure 7-30 in DOE, 2016]

DOE indicated that the safety function analyses demonstrate that the engineered and natural systems used to manage residual waste in WMA C are robust. DOE only observed moderate changes to the groundwater pathway dose results with respect to the margin between the base case and the performance standard.

In response to NRC's RAI 2-14 (NRC, 2019), DOE performed an additional analysis to look at the potential combined effect of additional uncertainties (DOE, 2019). DOE performed an analysis that examined the combined failure of multiple safety functions. DOE emphasized that they did not believe the analysis provided a credible representation of system behavior and that the analysis was not valuable for decision making. In the analysis, DOE set the infiltration rate to 5.2 mm/yr, assumed the basemat and infill grout behaved as sand to allow releases by advection, assumed that all waste was available for leaching, and set the aquifer dilution parameter to the 5th percentile value. For this case, the maximum dose was 0.078 mSv/yr (7.8 mrem/yr) at 640 years. DOE believed the results demonstrate the resiliency of the system.

3.9.2 NRC Evaluation of Performance Assessment Results and Protection of the Public

The NRC staff reviewed DOE's all pathway dose analysis results developed with deterministic, probabilistic uncertainty, and sensitivity analyses. The NRC staff considered all the different analyses when determining if DOE demonstrated compliance with 10 CFR Part 61.41.

One of the key aspects of PA analyses is the evaluation of the impacts of uncertainties. DOE's approach of using a deterministic base case, combined with probabilistic uncertainty analyses and safety function sensitivity analyses created challenges in getting a complete picture of the magnitude of the uncertainties involved. DOE's deterministic base case can be valuable for decision making when uncertainties are limited and well understood. However, a deterministic analysis that does not include the impact of uncertainties is of limited value for making safety-related decisions when there are numerous uncertainties, unless the deterministic analysis is demonstrably conservative (e.g., a screening analysis). Otherwise the impact of uncertainties is not being included in the decision-making process.

In the deterministic base case, DOE provided the result of a calculation, however, this result is not more or less likely than many other outcomes. The NRC staff describe, in previous sections of this TER, numerous parameters and models for the base case that were not adequately supported (e.g., basemat performance, advective flow in shrinkage gaps, aquifer dilution). For example, the actual effective diffusion coefficients for the basemat may be comparable to the value DOE assigned; however, they could also be orders of magnitude higher or lower. The actual values are unknown and have not been measured by DOE. When many parameters are not known, even if the ranges can be established through examination of other studies, it cannot be demonstrated that the base case "central tendency" performance is appropriate.

DOE supplemented the deterministic base case with probabilistic uncertainty analyses that were clearly described and used appropriate methods and approaches. The deterministic base case result was within the realizations of the probabilistic analysis. In the uncertainty analyses, DOE assigned some ranges to parameters that were too narrow (e.g., inventory, aquifer hydraulic conductivity) and did not include all sources of uncertainty. The NRC staff provided a list of some of the uncertainties in RAI 2-14 (NRC, 2019). Inclusion of broader ranges of uncertainty distributions would lead to more uncertainty in the output.

The safety function analyses were useful to demonstrate the significance of changes in safety functions to the outputs. The analyses demonstrated the resiliency of the outputs to changes to a <u>single</u> safety function. The system does appear to be resilient to the loss or deterioration of single safety functions. However, uncertainty and sensitivity analyses, including one-at-a-time sensitivity cases, do not identify risk-significant interdependencies and interrelationships between features and phenomena that could be evaluated with alternative scenarios or conceptual models. DOE developed the alternative conceptual models that were assessed independent of the safety function methodology and, partially, due to stakeholder interest. DOE did not analyze other potential plausible conceptual models that the NRC staff describes in the previous sections of this TER. For future PAs, DOE should document the results of the safety function methodology within their safety concept, thereby providing a technical basis for having included alternative conceptual models, or alternative scenarios, within the assessment and how they were deemed plausible. Also, documentation should be provided as to why potentially significant alternative conceptual models, or alternative scenarios (e.g., upstream Columbia River dams are decommissioned), were excluded from the assessment.

If the safety functions are mapped onto a matrix, DOE essentially evaluated the diagonal of the matrix. This approach is useful when there are limited uncertainties and the uncertainties are well-known, which is not the case for WMA C. The safety function analysis was used by DOE, and supplemented in their response to the RAI, in an attempt by DOE to account for uncertainties that were not included in the initial assessment. In contrast, a global uncertainty analysis would evaluate the full matrix and would assign, through probabilistic computations, the relative probability of occurrence of multiple conditions concurrently. Model uncertainty is extremely important and should be included by DOE in future probabilistic PA.

The case DOE analyzed in response to RAI 2-14 was informative and useful. DOE demonstrated the system would meet the performance objective with the significant deterioration of multiple safety functions. This case was an important source of information for the NRC staff to arrive at its conclusions. DOE described this case as an incredible or extreme scenario. While the assignment of essentially no performance of the infill grout is very pessimistic, and a variety of processes and barriers were not credited in the DOE evaluation, the sensitivity case evaluated in response to RAI 2-14 represents an unlikely, yet plausible, combination of events. The NRC staff disagrees that the case evaluated in response to RAI 2-14 represents an incredible or extreme scenario. As discussed in Section 3.3.2 of this TER, there are other studies that have long-term infiltration rates that are larger than DOE's maximum. As discussed in Section 3.5, DOE has yet to demonstrate that a grout can be implemented that will not result in a shrinkage gap that when combined with observed present-day in-leakage to tanks could result in advective release. Degradation of the tank walls with advective flow to waste on the periphery would not result in significantly different timing and magnitude of waste release than assumed in the DOE analysis. As discussed in Section 3.7, studies as recent as a few years ago had aquifer dilution values that were considerably lower than the minimum assigned by DOE.

Although the NRC staff has expressed technical challenges associated with each of the analysis techniques DOE used, overall the results demonstrate with reasonable expectation that DOE demonstrated compliance with 10 CFR Part 61.41 for WMA C. This conclusion applies to all tanks and ancillary equipment considered by DOE in the WMA C PA. DOE had a significant margin between their results and the performance objective.

3.9.3 NRC Conclusion on 10 CFR Part 61.41 Compliance

Based on the information provided, the NRC staff concludes that DOE has demonstrated compliance with 10 CFR Part 61.41 for all tanks and ancillary equipment included in the analyses for WMA C. This conclusion is based on the assumptions provided in this report and that future work by DOE validates these assumptions. The NRC staff provides recommendations with respect to the analyses supporting demonstration of compliance with 10 CFR Part 61.41 in the respective sections of this TER. Those recommendations do not change the conclusion with respect to compliance with 10 CFR Part 61.41. Those recommendations, if implemented, can enhance the technical basis for the Draft WIR Evaluation for WMA C, decrease uncertainties, or may be applicable to future waste determinations at other waste management areas, but are not essential for demonstrating compliance with 10 CFR Part 61.41.

3.10 Inadvertent Human Intrusion (10 CFR Part 61.42)

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. For the purposes of the analyses, and to be consistent with NRC guidance, DOE used an active institutional control period of 100 years. Inadvertent intruders are assuming to engage in normal behaviors associated with a rural lifestyle and consistent with current regional practices.

3.10.1 Summary of DOE Analyses of Inadvertent Human Intrusion

NRC's regulations for LLW disposal require the protection of individuals from inadvertent intrusion (10 CFR 61.42). DOE provides a similar requirement in Manual 435.1-1:

> "For purposes of establishing limits on the concentration of radionuclides that may be disposed of near-surface, the performance assessment shall include an assessment of impacts calculated for a hypothetical person assumed to inadvertently intrude for a temporary period into the low-level waste disposal facility. For intruder analyses, institutional controls shall be assumed to be effective in deterring intrusion for at least 100 years following closure. The intruder analyses shall use performance measures for chronic and acute exposure scenarios, respectively, of 100 millirem (1 mSv) in a year and 500 millirem (5 mSv) total effective dose equivalent excluding radon in air."

There are some minor differences between NRC's requirements and DOE's, but they are similar. DOE may consider inadvertent intrusion to be temporary in nature, whereas NRC's requirements do not explicitly address the permanence of the scenario. In practice, after the institutional control period, the site is assumed to be used for common activities under NRC's regulations. DOE's requirements allow the assumption of institutional controls for at least 100 years, whereas NRC's regulations allow a licensee to credit institutional controls for up to 100 years. In practice, an institutional control period of 100 years is assumed by commercial LLW disposal licensees. The primary difference is that DOE's requirements provide a dose limit of 1 mSv/yr (100 mrem/yr) for a chronic exposure scenario and 5 mSv/yr (500 mrem/yr) for an acute exposure scenario, whereas NRC's regulation currently does not provide a dose limit for inadvertent human intrusion. When NRC's regulations were developed a 5 mSv/yr (500 mrem/yr) whole body dose limit (or various organ dose limits) was used for inadvertent intruder analyses. In review of previous waste determinations and in guidance provided in NUREG-1854 (NRC, 2007), the staff has applied a 5 mSv (500 mrem) total effective dose equivalent limit for evaluation of inadvertent intrusion assessments.

3.10.1.1 Inadvertent Intrusion Assessment

DOE's inadvertent human intrusion assessment involved developing intruder exposure scenarios, compiling data and information for those exposure scenarios, and calculating the potential dose impacts to each receptor for the different potential sources of radioactivity remaining in WMA C after closure. If a robust barrier to intrusion was not present, DOE assumed that inadvertent intrusion occurred after the 100-year active institutional control period. If a robust barrier to intrusion was present, DOE assumed intrusion was delayed until 500 years after closure of WMA C.

In developing the intruder scenarios, DOE assumed that humans will 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. Two types of exposure scenarios were considered in the WMA C PA to estimate dose to the hypothetical intruder: acute exposure scenarios and chronic exposure scenarios. Acute scenarios evaluated the dose received from drilling a well and subsequent exposure to residual waste in the drill cuttings; acute exposure is evaluated over a short time period. Chronic scenarios evaluated the dose received from spreading the drill cuttings over a specific area while living and/or working on that area. DOE evaluated one acute exposure scenario and three chronic exposure scenarios in the WMA C PA.

The intruder scenarios were initially applied to each of 19 different waste sources (twelve 100-series tanks, four 200-series tanks, the CR Vault, catch tank C-301, and pipelines). The waste present in each source was assumed to be uniformly distributed (i.e., uniform thickness) over the total plan-view area of the source. The average concentration of radioactivity in each waste source was used based on both the average radioactivity per unit mass or volume and the average thickness of waste over the source. DOE considered the decay and ingrowth of radioactivity but did not consider any depletion due to transport of radionuclides out of the systems. In the Draft WIR Evaluation for WMA C, DOE indicated that intruder doses for the pipelines were based on the waste transfer pipelines that are 3 in (7.6 cm) diameter and assumed to be 5 percent full of waste because those were the most common pipeline type and size. Evaluation of a fully plugged cascade pipeline was considered in a sensitivity analysis.

DOE discusses the probability of intrusion in the WMA C PA (Section 9.1), including a qualitative discussion on the likelihood of inadvertent human intrusion at WMA C. The information was provided to be contextual, but the likelihood of intrusion was not explicitly considered in comparing the intrusion analysis results to regulatory standards. DOE stepped through the series of events that would need to occur for inadvertent intrusion into waste residuals to occur at WMA C. Presently, the site access is controlled; active institutional controls are in place to ensure that a member of the public will not inadvertently intrude. Therefore, for inadvertent intrusion to occur, first, active institutional controls would need to lapse or be eliminated. Next, societal memory of the existence or the location of buried radioactivity would need to be lost. Third, someone would need to drill for resources (e.g., water) and not have a desire to obtain those resources from a different source, such as the Columbia River. Next, the drilling would need to occur at the waste location. Finally, the drill would need to penetrate the waste and the driller would need to not recognize that something atypical (and dangerous) has been encountered. DOE concluded that the likelihood of drilling through a tank at WMA C is very small; the likelihood of drilling through a pipeline is also small but larger than that of drilling through a tank.

The NRC staff requested additional information on the impacts to inadvertent intruders from plugged pipelines and from intrusion into tank TK-CR-011 in the CR Vault (NRC, 2019). In response to the RAI, DOE provided the results of additional intruder dose calculations (DOE, 2019).

3.10.1.2 Intruder Protection Features

DOE's closure plans for the systems in WMA C are designed to stabilize the waste residuals and protect public health and safety. The existing systems in which waste residuals are present are comprised of steel, concrete, or both. Those systems will be covered by multiple meter thick engineered surface cover. The tanks and CR-Vault will be grouted prior to closure. The formulation and properties of the grout have not yet been finalized. DOE indicated that the engineered materials are expected to degrade over time but that they would likely be able to provide a deterrence to intrusion for some period of time. The pipelines will not be grouted; therefore, DOE assumed the steel would not provide a barrier to intrusion by drilling starting at 100 years after closure. For the systems that would be grouted, the combination of steel, reinforced concrete, and grout was assumed to provide a barrier to intrusion for 500 years after closure.

DOE accounted for the depth to waste in the intruder assessment. Most of the residual waste is present at depths below the land surface greater than 3 meters (10 ft) in the present day. For example, the bottoms of the 100-series tanks are approximately 11.5 m (38 ft) below the existing land surface. Pits and the tops of some tanks are within 3 meters of the existing land surface without a closure cover present. All the waste residuals will be at depths greater than 3 meters (10 ft) after installation of the closure cover when waste retrieval is complete. If the depth to waste is large enough (e.g., greater than 10 ft [3 m]), the probability of future excavation into the waste can be greatly reduced. In addition, the disposal of waste at greater depths can decrease the likelihood of exposure of the waste by other means, such as erosion.

3.10.1.3 Exposure Scenarios in the Intruder Analysis

Because of the depth to waste, DOE did not evaluate potential excavation into the waste residuals. DOE evaluated a short-term exposure (acute well driller exposure scenario) of a well driller to drill cuttings that are exhumed from a well installed to supply water. The depth of the well was the depth of the water table (79 m [260 ft]). DOE also evaluated chronic exposure to the drill cuttings for someone who resides at the site. For the acute exposure scenario, the concentration of radioactivity in the cuttings was not assumed to be diluted by mixing with clean soil. For the chronic exposure scenarios (e.g., rural pasture [5,000 m^2 (54,000 ft^2)], suburban garden [2,500 m^2 (27,000 ft^2)], commercial farm [647,000 m^2 (7,000,000 ft^2)], the drill cuttings were assumed to be distributed over the area of the property and mixed uniformly to a depth of 15 cm (6 in). For the suburban garden exposure scenario, the size of the garden was 100 m^2 (1,100 ft^2), large enough to supply 25 percent of the daily vegetable intake for a family of four. Table 3-15 provides a description of the intruder scenarios evaluated in the WMA C PA. The exposure pathways differ for the different chronic exposure scenarios. DOE assumed that contaminated water would not be used by receptors in the chronic inadvertent intruder exposure scenarios.

Table 3-15 Inadvertent Intruder Scenarios for the WMA C Performance Assessment

Exposure scenario	Description
Acute Exposure: Well Driller	Dose is the result of drilling through WMA C. Exposure pathways include external exposure, inhalation of soil particulates, and incidental soil ingestion. Exposure occurs during the drilling operation while in contact with the drill cuttings. Exposure does not depend on the borehole diameter but depends on the thickness of the waste.
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 pathways 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 pathways 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 pathways are external exposure, inhalation of soil particulates, and incidental soil ingestion.

3.10.1.4 Protection of Individuals from Inadvertent Intrusion Results

The calculated doses to an inadvertent intruder are presented in Table 3-16. The largest dose DOE calculated in the Draft WIR Evaluation for WMA C was for the acute intruder from the pipeline source at 0.36 mSv (36 mrem), or 7.2 percent of the performance objective (5 mSv [500 mrem]). By comparison, the largest dose for the chronic intruder was 0.082 mSv (8.2 mrem) for intrusion into the pipelines. Though DOE's performance objective for the chronic intruder is lower (1 mSv [100 mrem]) compared to the acute intruder (5 mSv [500 mrem]), the chronic doses DOE calculated were all much lower than the acute doses because of the assumed tilling depth of 15 cm (6 in). For a 16.5 cm (6.5 in) diameter well, approximately 1.7 m^3 (60 ft^3) of soil/waste mixture is extracted as cuttings. For the suburban gardener, the mixing volume that these cuttings are distributed in is approximately 375 m^3 (13,200 ft^3). The exposure time for the chronic receptor is significantly longer than the exposure time for the acute receptor, however, the concentrations of radioactivity are significantly lower, resulting in comparable overall results for the acute and chronic exposure scenarios.

DOE calculated doses for the compliance period (1,000 years) and for the sensitivity/uncertainty analysis period (1,000 to 10,000 years). The acute exposure scenario doses are dominated by ^{137}Cs and ^{239}Pu, while the chronic exposure scenario doses are dominated by ^{90}Sr, ^{137}Cs, and ^{239}Pu. The intruder doses for the sensitivity/uncertainty analysis period are lower than the doses for the compliance period, as a result of radioactive decay.

Table 3-16 Dose Results for the Inadvertent Intruder Scenarios

Source	Acute Dose (mrem/yr)	Chronic Commercial Farm Dose (mrem/yr)	Chronic Rural Pasture Dose (mrem/yr)	Chronic Suburban Gardener Dose (mrem/yr)
241-C-101	1.24	2.17×10^{-3}	0.144	0.322
241-C-102	4.59	8.09×10^{-3}	0.537	1.20
241-C-103	0.409	7.25×10^{-4}	6.14×10^{-2}	0.11
241-C-104	0.577	1.10×10^{-3}	0.121	0.17
241-C-105	3.80	6.69×10^{-3}	0.718	1.23
241-C-106	3.47	8.75×10^{-3}	0.893	0.957
241-C-107	14.9	2.66×10^{-2}	1.82	3.90
241-C-108	5.80×10^{-2}	1.05×10^{-4}	1.09×10^{-2}	1.71×10^{-2}
241-C-109	3.10×10^{-2}	5.57×10^{-5}	7.63×10^{-3}	9.33×10^{-3}
241-C-110	8.24×10^{-2}	1.78×10^{-4}	1.99×10^{-2}	2.44×10^{-2}
241-C-111	7.47	1.32×10^{-2}	1.40	2.13
241-C-112	0.348	6.10×10^{-4}	9.17×10^{-2}	0.141
241-C-201	14.5	2.52×10^{-2}	1.58	3.75
241-C-202	12.8	2.22×10^{-2}	1.39	3.32
241-C-203	0.461	8.51×10^{-4}	7.25×10^{-2}	0.126
241-C-204	5.60×10^{-2}	1.77×10^{-4}	2.97×10^{-2}	2.49×10^{-2}
241-C-301	21.2	3.86×10^{-2}	2.69	5.57
CR-VAULT	3.91	7.10×10^{-3}	0.496	1.03
Pipeline	36.0	1.13×10^{-3}	8.21	3.92

In response to the NRC staff's RAI, DOE also calculated intruder impacts (chronic) to a plugged pipeline, the CR Vault, and tank CR-011 within the CR Vault. DOE calculated the acute intruder dose of 0.36 mSv (36 mrem) for intrusion into a 100 percent plugged pipeline. The corresponding chronic intruder dose was 1.6 mSv (160 mrem) assuming closure in year 2020 and loss of institutional control at year 2120. DOE performed a sensitivity case to examine the impacts from a later closure date. Using a closure date of 2068, the peak chronic dose for a 100% full pipeline was 0.5 mSv (50 mrem). DOE indicated that diversion boxes and pits are expected to contain no measurable volume of waste, therefore, would not provide more limiting dose results to the intruder. The acute peak dose for the CR Vault was 0.039 mSv (3.9 mrem) and for the CR-011 tank the acute peak dose was 0.44 mSv (44.2 mrem) at 500 years after closure. DOE revised the results for a normal pipeline (5% full of waste – not plugged) to 0.018 mSv (1.8 mrem) for the acute intruder from the initially calculated 0.36 mSv (36 mrem) in the Draft WIR Evaluation.

3.10.2 NRC Evaluation of Inadvertent Human Intrusion

The NRC staff evaluated DOE's assessment of inadvertent human intrusion by reviewing the Draft WIR Evaluation for WMA C, the WMA C PA, supporting documents, and GoldSim model files. The NRC staff reviewed the requirements applied, DOE's approach to the assessment, intruder protection systems, exposure scenarios, parameters, models, assumptions, and results of the assessment. DOE took credit for three primary features with respect to intruder protection: active institutional controls, depth to waste, and engineered components and materials. Additional features were discussed and may provide defense-in-depth.

3.10.2.1 NRC Evaluation of Inadvertent Intrusion Assessment

The regulatory criteria DOE applied were appropriate for the assessment. DOE has a more restrictive dose limit for chronic intruder scenarios of 1 mSv (100 mrem) per year compared to NRC's 5 mSv (500 mrem) per year. This application of a more restrictive limit is protective of public health and safety. DOE applied an active institutional control period of 100 years from an assumed closure date of 2020. These assumptions with respect to institutional controls are consistent with NRC regulations and guidance. Because of the additional work such as regulatory approvals, development and installation of an engineered cover, final waste removal and sampling, and grout formulation development and implementation, DOE's use of a 2020 closure date for WMA C is conservative. Because much of the intruder doses are driven by short-lived isotopes (e.g., ^{137}Cs and ^{90}Sr), the intruder doses are conservative with respect to the assumed closure date (i.e., allowing for less radioactive decay than expected). The assumed institutional control period is likely to be conservative because the active institutional control period will not begin until the 200 East area is remediated. DOE anticipated that remediation will not be completed for many decades after the assumed date of 2020. For NRC's commercial licensees, financial assurances for institutional controls must be provided to ensure the active institutional controls are funded and can be implemented. DOE's funding for active control of disposal sites is determined by annual budget appropriations after operations are completed, which is appropriate. The period of time that DOE is active at the Hanford Site will be determined by the overall progress at the site and appears to be a minimum of multiple decades. DOE's use of a 100-year active institutional control period is appropriate and consistent with NRC requirements.

DOE discussed the probability of intrusion in the Draft WIR Evaluation and the WMA C PA. DOE indicated that while the potential doses that might arise from intrusion into a tank are higher than from intrusion into a pipeline (non-plugged), the likelihood of occurrence of intrusion into a tank is very small ("vanishingly small"). In the WMA C PA, DOE stated the intrusion analyses for tanks are informational. DOE did not incorporate probability estimates into the comparison of the results with the performance objectives. DOE's qualitative evaluation of intrusion probability is highly speculative and based on many unstated assumptions. The probability of someone disturbing waste at the site in the future is a complex function of future land use, socioeconomic conditions and development, technology evolution, and material degradation and performance. Probability values can be estimated but the uncertainty range is so large as to make the results not useful for regulatory decision-making. DOE considered different components to estimating intrusion probability. NRC review of these components are provided in the list below:

- Loss of Institutional Control – Active institutional controls will be effective as long as they are funded and implemented. Funding is likely to decrease or be eliminated when perceived risks are eliminated. The NRC staff concurs with DOE that the likelihood at 100 years is lower than the likelihood at 1,000 years. Given the amount of time remediation is going to take at the 200 East area a 100-year time period from 2020 is likely to be conservative. However, there is no precedent for the effectiveness of land use control over such long timeframes.

- Loss of Societal Memory – Societal memory will deteriorate over time. Many people do not know much about their great-grandparents (~100 years) and few companies survive for more than 100 years. Electronic media and records change rapidly such that without a dedicated effort to preserve information, it can be lost. DOE and its contractors have developed over 4 million documents about the Hanford Site, most of which are difficult for a member of the public to access. Despite DOE's records of where materials and structures are located, the NRC staff determined upon review of incident reports that multiple instances have occurred where DOE accidentally disturbed pipelines. Disturbances would be even more likely for an inadvertent intruder without access to these records.
- Decision to drill on the Central Plateau – DOE believed that the probability that someone would drill on the Central Plateau is low, indefinitely, because activity would be more likely to be closer to the river. Figure 3-32 is a snapshot of current wells from the State of Washington Department of Ecology. It shows numerous wells in the vicinity of the Hanford Site and there does not appear to be a correlation between the well location and the distance from the river. While water may be obtained from the river, the decision to install a well or obtain river water is likely to be purely an individual financial decision.
- Decision to drill at WMA C – The decision to drill at WMA C specifically is likely to be random based on property ownership and land use. The WMA C area is approximately 22,500 m^2 (242,000 ft^2) and the area of contamination is approximately 7,500 m^2 (81,000 ft^2) for waste residuals (all source types), or approximately a 1 in 3 chance of encountering contamination. This is per drilling event and would need to be integrated over time for the drilling frequency per unit area. The NRC staff does not agree with DOE that the probability of drilling at WMA C is low to extremely low over the timeframes considered.
- Penetration of drill bit into waste – DOE indicated the probability for pipelines is moderate to high and for tanks is extremely low, in perpetuity. At 100+ years, exposed carbon steel is not likely to provide a substantial barrier to modern drilling technologies. DOE's grout formulation for backfilling the tanks and structures has not yet been established, and rheology concerns with filling all void spaces may necessitate a high water-to-cement ratio formulation that would generally have lower mechanical strength. Reinforced concrete would be expected to provide a barrier to drilling depending on the drilling technology and regional practices. Some of the buried structures are heavily reinforced. The ability of fresh or aged reinforced concrete to withstand drilling is an open technical question. NRC is conducting research into this question by surveying drillers about their practices. Other programs (such as the IAEA in their borehole disposal program for disused sources) make use of deflector plates of high strength steel placed at an angle. However, this would not be practical given the geometry of the residual waste at WMA C. While a thick layer of concrete with thin layers of carbon steel would undoubtedly make drilling more difficult, whether it would prevent drilling and for what period of time is an open technical question. The concrete domes of the tanks have high-strength concrete and a very large amount of rebar.

- Experience of the driller – DOE indicated that the driller would recognize something different upon intrusion and stop drilling. They stated that for pipelines it is a moderate to high probability that the driller would not recognize something different but for tanks it would be extremely low probability. Based on preliminary results of NRC's surveys with drillers, this does not appear to be the case. Drillers indicated they would communicate with the client when drilling became much more difficult than expected but if the client was willing to pay drilling would continue.

The NRC staff recognizes that DOE did not credit intrusion probability in their Draft WIR Evaluation, but by including the language from the PA about the probability of intrusion, a mixed message is provided to stakeholders. The NRC staff does not agree that intruder drilling probabilities are vanishingly small or that the results for drilling through a tank 500 years in the future are not meaningful for comparing to performance objectives and making regulatory decisions. The higher dose limits applied to intruders compared to an offsite member of the public already incorporates a probability or likelihood of future site use. For NRC's intruder requirements, the temporal-integrated probability over the time that the waste remains hazardous is 5 percent. NRC expects that the active institutional controls that are put in place will be effective beyond 100 years, however, that the effectiveness cannot be ensured. Waste buried more deeply and with more engineered barriers will have a lower likelihood of being disturbed because less actions are taken that disturb material deeper underground.

Figure 3-33 is a plan view map of the Waste-Isolation-Pilot-Plant (WIPP) land withdrawal act area showing nearby oil and gas wells. The map shows the effectiveness of robust controls (e.g., a federal law with continual compliance oversight). There are numerous wells right on the boundary of the excluded area. If this was a map of deep wells from 100 years ago, there would be no black dots on the figure. This should stress the importance and large influence of socioeconomic and technology changes on intruder drilling probability.

3.10.2.2 NRC Evaluation of Intruder Protection Features

An intruder protection system that DOE took credit for is the depth to waste. Intruder excavation exposure scenarios tend to have much larger dose impacts compared to intruder drilling exposure scenarios because much more waste is exhumed. As discussed in Section 3.5, most of the residual waste remaining in the tanks is present in layers on the bottom of the tanks or adhered to the tank walls. The tank bottoms and walls are more than 3 m below the present-day land surface and in some cases much deeper. DOE plans to implement a closure cover that will ensure all residual wastes are more than 3 m below the land surface at the time of closure. The NRC staff agrees with DOE's elimination of intruder excavation exposure scenarios based on the depth to waste, conditional on the assumption of installation of a closure cover of appropriate thickness.

DOE described the engineered materials that comprise the systems in WMA C and how they may impact future inadvertent intruders. For waste residuals in the carbon-steel pipelines, DOE did not credit the pipelines themselves as affording a barrier to intrusion. This is reasonable considering the potential corrosion of the pipelines during the 100-year active institutional control period. DOE took credit for the engineered materials used to construct the other systems in which waste residuals are anticipated to remain at closure.

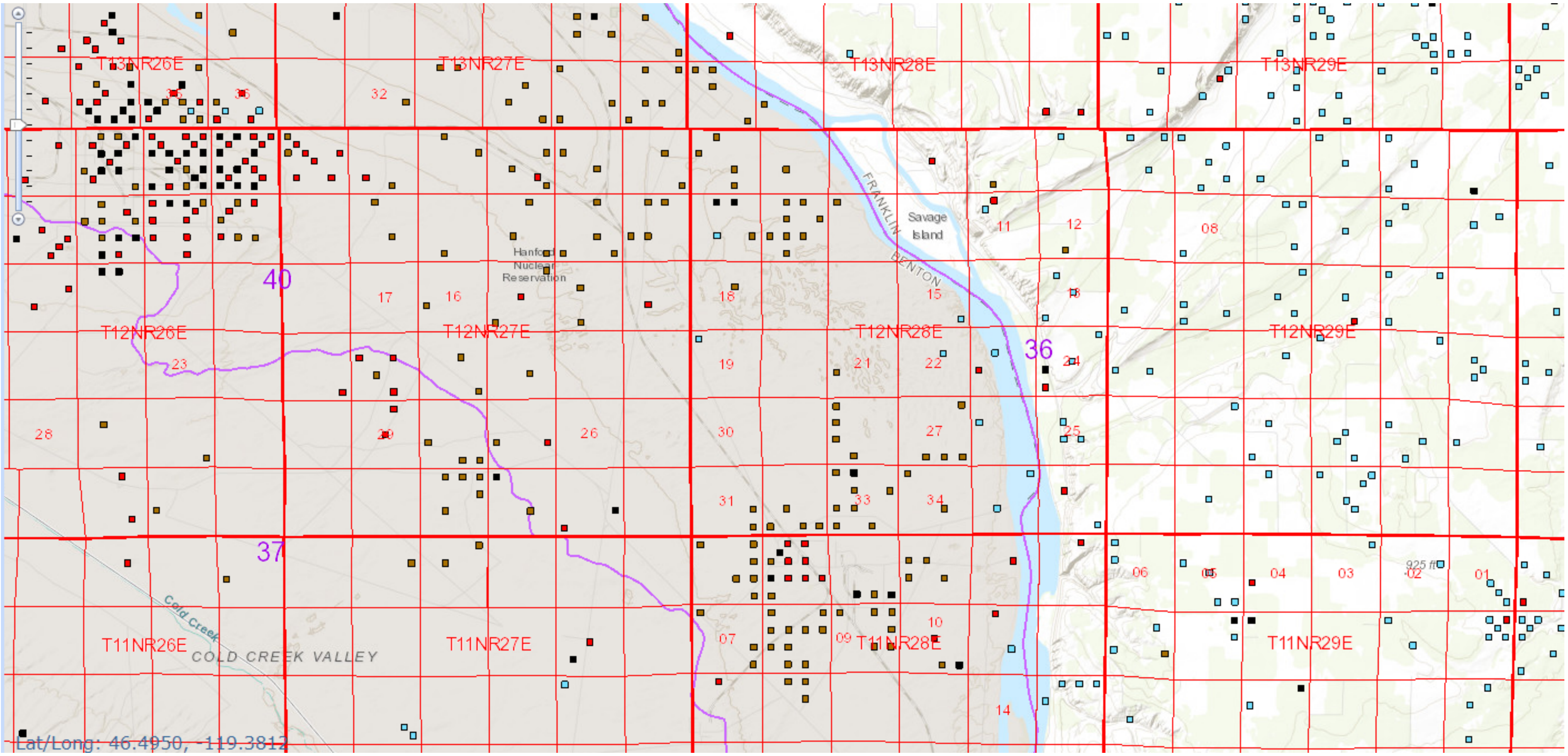


Figure 3-32 **Wells in the Vicinity of the Hanford Reservation**
(Generated from WA Department of Ecology well construction and licensing search tool https://appswr.ecology.wa.gov/wellconstruction/map/WCLSWebMap/default.aspx)

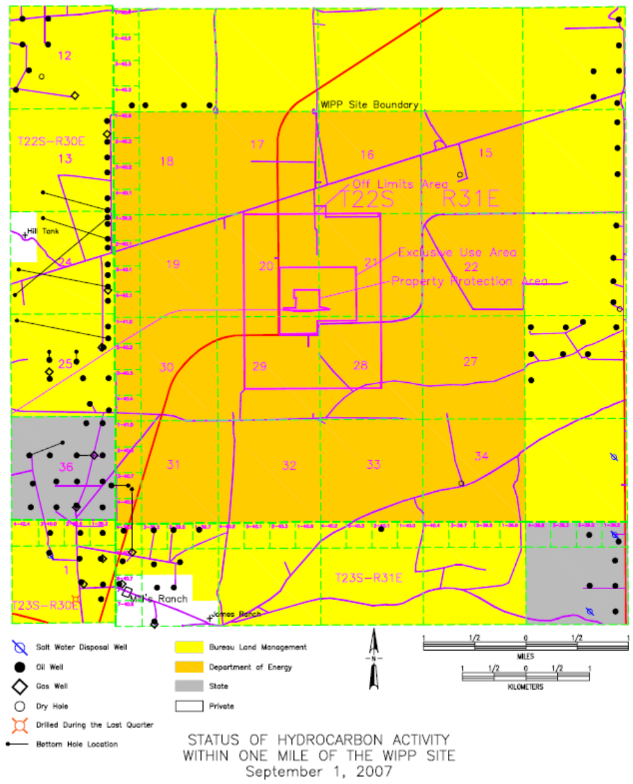


Figure 3-33 WIPP Land Withdrawal Act Area and Nearby Wells
(DOE/WIPP-10-2308, 2010)

DOE indicated that the engineered materials would be anticipated to degrade over time but that they would be expected to provide a deterrence to intrusion for some period of time. For the systems that would be grouted, DOE assumed the combination of steel, reinforced concrete, and grout would provide a barrier to intrusion for 500 years after closure. DOE provided engineering judgment as the basis that the combination of steel, reinforced concrete, and grout would provide 500 years of delay in potential drilling after closure. As shown in Section 3.10.2.3, the timing of potential intrusion into a tank is one of the most important variables because most of the dose comes from ^{137}Cs and ^{90}Sr, which experience significant reductions over 500 years (roughly an order of magnitude for each 100 years). For example, if someone were to drill into Tank C-111 at 100 years, the acute intruder dose would be 2.06 mSv (205.6 mrem) whereas the chronic intruder dose for the rural pasture exposure scenario would be 7.1 mSv (7,135 mrem). It would take until approximately 278 years (178 years after the end of the active institutional control period) for the dose to decrease below the DOE requirement of 1 mSv (100 mrem) per year. The NRC staff do not believe there is quantitative technical information as to the effectiveness of the types of engineered components associated with the 100- and 200-series tanks to deter modern drilling technologies from penetrating them for 100's of years in the future. The NRC staff does agree it is reasonable to assume these materials will deter drilling for enough time that the radioactivity will decay. However, it is recommended that a quantitative basis be developed for this key assumption (Recommendation #43). It would not be cost prohibitive to complete studies for drilling (with knowledge and without knowledge of the barrier presence) through buried engineered materials; the information may provide considerable cost savings from a risk-informed standpoint with respect to future waste retrievals and the closure of other waste management areas.

The tanks, catch tank, and CR-Vault will be grouted prior to closure. DOE has not yet finalized the formulation and properties of the grout. The mechanical properties of the grout may vary substantially depending on the grout formulation. Access points to emplace grout may be limited in some systems, therefore, rheological considerations may drive the grout formulation. High water-to-cement ratios may be needed to ensure proper flow and filling of the system, but these cements tend to have lower strength and may not provide a deterrence to drilling. It is assumed that DOE will identify the design requirements for grout with respect to intruder protection and that this design will be implemented for closure. DOE will need to balance the requirements of the grout with respect to the different performance objectives.

3.10.2.3 NRC Evaluation of Exposure Scenarios in the Intruder Analysis

The NRC staff reviewed DOE's exposure scenarios, exposure pathways, and intruder dose assessment parameters and assumptions. Because of the depth to waste, DOE did not evaluate potential excavation into the waste residuals. The NRC staff evaluated the information DOE provided for the depth to waste residuals and concur with the elimination of the intruder excavation exposure scenario. DOE evaluated short-term (acute well driller exposure scenario) and long-term (chronic exposure scenarios – suburban gardener, rural pasture, and commercial farm) exposure scenarios resulting from installation of a well and exposure to contaminated drill cuttings. The exposure scenarios included pathways for inadvertent soil ingestion, inhalation, and exposure to external radiation. The suburban gardener exposure scenario included the vegetable consumption pathway and the rural pasture exposure scenario included the milk consumption pathway. Figure 3-34 provides the exposure pathways for the inadvertent intruders by eliminating those pathways that were considered for a member of the public for offsite contamination. DOE clearly explained which pathways were included for different intruder scenarios, but did not explain why the pathways for the intruder scenarios were defined differently than for offsite receptors (i.e., releases) in the PA. Pathways were eliminated for the intruders that were included for the onsite receptors.

Figure 3-35 shows the intruder doses for the suburban gardener and rural pasture (chronic) receptors calculated by the NRC staff from DOE's GoldSim model results. The plant ingestion and milk ingestion pathways are the largest contributors to the dose, much larger than the sum of all other pathways. The importance of the institutional control period and the engineered materials that provide a barrier to intrusion and delay contact with the radioactive material is evident. The long-term doses decay such that the doses are below 1 mSv/yr (100 mrem/yr) by 200 years and are approximately 0.02 to 0.04 mSv (2 to 4 mrem) by 500 years, a small fraction of the performance standard.

For the chronic exposure scenarios, when the drill cuttings are brought to the land surface, they are assumed to be spread over the whole area of the property and then tilled to depth of 15 cm (6 in) (e.g., rural pasture [5,000 m^2 (54,000 ft^2)], suburban garden [2,500 m^2 (27,000 ft^2)], commercial farm [647,000 m^2 (7,000,000 ft^2)]). The volume of cuttings is small, such that a very large dilution factor to the concentration of radioactivity results.

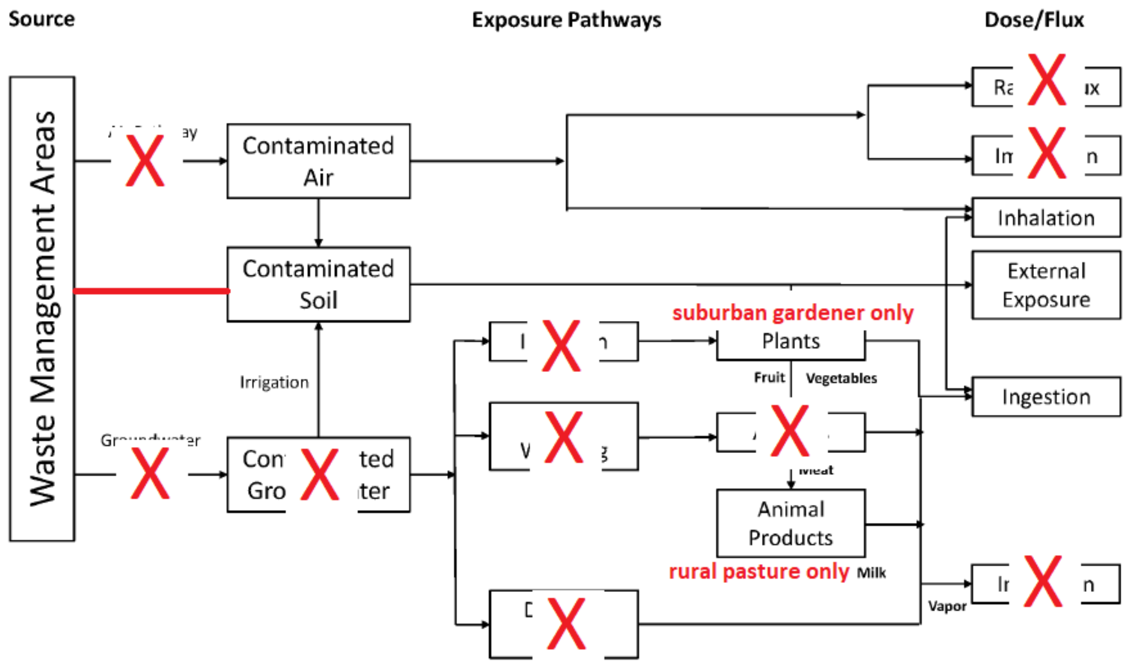


Figure 3-34 Exposure Pathways for the Inadvertent Intruders

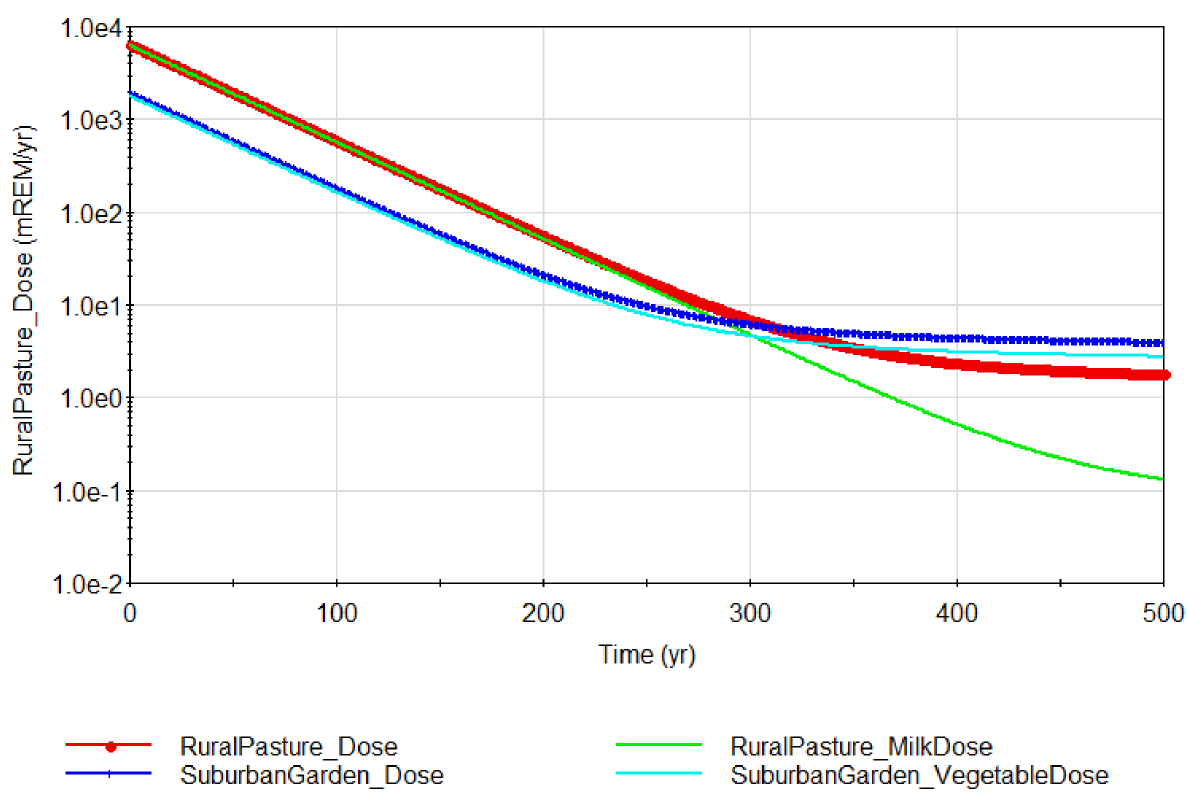


Figure 3-35 Plant and Milk Pathway Doses for Tank C-107

The assumed tilling depth is likely only to be reasonable for application to those areas of the property that are actively manipulated for growing crops, such as a garden for the suburban garden exposure scenario or the pasture for the rural pasture exposure scenario. For the inhalation pathway, the concentration of radioactivity in soil may be much higher if the assumed mixing depth for the cuttings is smaller and consistent with natural phenomenon. This assumption would impact calculated doses for the external exposure, inadvertent soil ingestion, and inhalation pathways. It would have a lesser impact on plant and milk ingestion because the radioactivity must be transferred from the soil to the plants and much of this occurs from root uptake. For a resident exposure scenario, it would be appropriate to use the 15 cm (6 in) mixing depth for the garden and a smaller value for the remainder of the property. It is recommended that DOE reconsider the mixing assumptions for future evaluations to ensure the assumed mixing depths are consistent with projected land use for the chronic intruder scenarios (Recommendation #44). Less mixing would increase the contribution of the external exposure, inadvertent soil ingestion, and inhalation pathways but the increase would not be substantial enough to change the conclusions of the Draft WIR Evaluation for WMA C.

DOE provided vegetable consumption data in the PA, Table 6-22 and Table 9-5 (DOE, 2016). DOE used different approaches for the offsite receptor and inadvertent intruders. For the offsite receptor, DOE used a “representative person” (which according to DOE-STD-1196-2011 (2011) is required to be age- and gender-weighted), then used the 95th percentile of the underlying distribution. The locally-grown fruit and vegetable consumption rate was 68 kg/yr (150 lb/yr) for the offsite receptor. For the onsite receptor, the locally-grown fruit and vegetable consumption rate was 26.6 kg/yr (59 lb/yr). The onsite receptor was not required to be a representative person. For the onsite receptor, DOE evaluated the amount of produce that could be raised from a 100 m^2 (1,100 ft^2) garden, and determined that it would be enough to supply the annual needs of an individual, but only about 25 percent of a family of 4 (one of two calculations produced an area of 67 m^2 (720 ft^2) to supply the fruit and vegetable needs for an individual) (HNF-SD-WM-TI-707 Rev 4, 2004).

The fruit and vegetable consumption values used by DOE for the onsite receptor represent average population values more than values appropriate for a gardener. The selection of family size and garden size are clearly documented but somewhat arbitrary. If the receptor is someone who desires to produce homegrown fruits and vegetables, they may size their garden to produce the needs of themselves (and their family). For comparison, NUREG/CR-5512, Volume 3, Table 6.21 provides ingestion rates for homegrown foods (NRC, 1999). The 50th percentile value for ingestion of homegrown vegetables and fruits (not grains) is 59 kg/yr (130 lb/yr), and the 95th percentile is 401.65 kg/yr (884 lb/yr) (over 2 pounds (1 kilogram) per person per day). The 50th percentile value (without the assumed 25 percent locally grown fraction) is consistent with the value DOE used for the offsite receptor. The intruder need not be “worst case” maximizing the intruder location, scenario and parameterization. However, the intruder is a hypothetical construct designed to ensure protection from the inadvertent use of the disposal site into the distant future that could include activities that are less likely but more disruptive. In future waste evaluations it is recommended that DOE use consistent approaches for the onsite and offsite receptors with respect to fruit and vegetable ingestion (Recommendation #45).

DOE used a mass loading factor of 66.6 μg/m^3 (4.2x10^{-9} lb/ft^3) for all receptors (onsite, offsite, acute, chronic) (DOE, 2016). The source of the value is NCRP Report No. 129 (NCRP, 1999). Mass loading values are important to calculate the impacts from inadvertent inhalation of contaminated soil that is resuspended in the atmosphere. Detailed mass loading values and resuspension studies have been completed at the Hanford Site (Sehmel, 1977). Those studies provided a range for long-term respirable soil mass loading of 7 $\mu g/m^3$ to 700 $\mu g/m^3$ ($4.4x10^{-10}$ to $4.4x10^{-8}$ lb/ft^3). Site-specific values for biosphere parameters should be used when available (Recommendation #46). Mass loading is particularly sensitive to atmospheric conditions (e.g., wind speed, moisture conditions). Mass loading is also sensitive to activity levels and disturbance processes. It was not clear to the NRC staff why the value prescribed for the acute driller was set equal to the long-term atmospheric value. The NRC staff recommends that DOE measure mass loading values that can be assigned to an acute intruder (well driller) to support future PAs (Recommendation #47). Drilling occurs at the Hanford Site with some regularity, therefore, there is opportunity to collect site-specific data. The acute drilling dose to an intruder through a cascade pipeline was 0.36 mSv (36 mrem) at 100 years, of which 0.21 mSv (21.1 mrem) was attributed to external exposure, 0.11 mSv (10.6 mrem) to inhalation, and the remainder to inadvertent soil ingestion. An increase in soil mass loading would have a corresponding direct increase to the inhalation dose to the intruder. The acute intruder doses would meet the performance objective (5 mSv [500 mrem]) even with an order of magnitude increase in mass loading values.

DOE used a separate performance standard for radon by prescribing a flux limit rather than incorporating the dose impacts from radon into the dose assessment for the intruder. NRC does not explicitly list in the regulation (10 CFR Part 61) how radon should be accounted for because at the time the regulation was developed only minor amounts of uranium-bearing waste were expected to be disposed as LLW. The NRC staff evaluated this issue as part of potential revisions to 10 CFR Part 61 to provide criteria for the disposal of large amounts of depleted uranium. Staff concluded that NRC never intended for radon to be excluded from the dose contributions from other radionuclides. The problem with prescribing a flux limit is that indoor radon doses are roughly an order of magnitude higher than outdoor doses and the decrease of flux rate to the land surface as compared to a home basement can be substantial. In addition, DOE has some sources that have very high amounts of uranium (e.g., Tank C-203 has over 50 wt.% uranium) that over time could produce significant amount of radon. DOE indicated that the radon flux from Tank C-203 was estimated as $7x10^{-3}$ pCi/m^2-s, which is well below the 20 pCi/m^2-s standard, but this result is for a point at the land surface that may not be representative of the result for the home of an inadvertent intruder. The NRC staff recommends that DOE include radon with the dose impacts to the inadvertent intruder in future waste evaluations to increase transparency with stakeholders and better communicate the significance of managing concentrated uranium-bearing waste (Recommendation #48).

3.10.3 NRC Conclusion on 10 CFR Part 61.42 Compliance

The NRC staff evaluated DOE's Draft WIR Evaluation for WMA C to determine if the residual waste will be managed to meet safety requirements comparable to the performance objectives set out in 10 CFR 61 Subpart C. Protection of inadvertent intruders is provided by 10 CFR 61.42. As shown in Table 3-16, DOE's calculated intruder doses all were below the 5 mSv (500 mrem) acute and 1 mSv (100 mrem) chronic standards except for the dose resulting from intrusion into a plugged pipeline that was 100% full of waste. The chronic dose using an assumed 2020 closure date was 1.6 mSv (160 mrem) and was 0.5 mSv (50 mrem) using an assumed closure date of 2068. The results show the impact of the closure date as the short-lived activity is the biggest component of the initial intruder doses. Based on current progress for remediation of WMA C and the 200 East area, the 2020 date is conservative and the 2068 date or something comparable is more appropriate.

The NRC staff concludes that DOE has demonstrated compliance with 10 CFR 61.42 for the tanks and ancillary equipment except the plugged pipeline(s). Though DOE made the most conservative estimate possible assuming the total pipe volume being full of waste, the radiological composition of the waste remaining in a plugged pipeline is highly uncertain; it could be orders of magnitude more or less than assumed (see Section 3.5.2.2). DOE did not indicate that they have plans to characterize the plugged pipeline(s) to verify their inventory assumptions. The phenomena that results in plugging is very complex, and the records are incomplete to have confidence in the inventory assumptions without analytical characterization or other means of verification.

3.11 Protection of Individuals During Operations (10 CFR Part 61.43)

The performance objective in 10 CFR 61.43, protection of individuals during operations, states the following:

> Operations at the land disposal facility must be conducted in compliance with the standards for radiation protection set out in part 20 of this chapter, except for releases of radioactivity in effluents from the land disposal facility, which shall be governed by §61.41 of this part. Every reasonable effort shall be made to maintain radiation exposures as low as is reasonably achievable.

3.11.1 Summary of DOE Results for Protection of Individuals During Operations (10 CFR Part 61.43)

Section I.E(13) in DOE Manual 435.1-1 (DOE, 2007) is a provision that provides for protection of individuals during operations and comparable to 10 CFR 61.43:

> “Radioactive waste management facilities, operations, and activities shall meet the requirements of 10 CFR Part 835, Occupational Radiation Protection, and DOE [Order] 5400.5 [now DOE Order 458.1], Radiation Protection of the Public and the Environment.”

This requirement references 10 CFR Part 20, “Standards for Protection Against Radiation”, which contains radiological protection standards for workers and the public. 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.

In the Draft WIR Evaluation for WMA C, DOE provided a crosswalk of the relevant DOE regulation or limit consistent with that provided in 10 CFR Part 20 to demonstrate that the DOE regulation provides an equivalent level of protection. The cross-referenced “standards for radiation protection” in 10 CFR Part 20 that are considered in the Draft WIR Evaluation for WMA C are the dose limits for the public and the workers during disposal operations set forth in 10 CFR Part 20, Subpart B—Radiation Protection Programs (see Table 3-17). The Draft WIR Evaluation for WMA C addressed the dose limits for the public and workers during disposal operations set forth in 10 CFR Part 20, and similar provisions in DOE regulations and Orders.

Table 3-17 Crosswalk Between DOE Requirements and the Relevant Standards Set Forth in 10 CFR 20. [Table 5-4 in DOE, 2018]

10 CFR 20 Standard	DOE Requirement	Basis Document Section	Title
10 CFR 20.1101(d)	DOE Order 458.1	5.4.1	Air Emissions Limit for Individual Member of the Public
10 CFR 20.1201(a)(1)(i)	10 CFR 835.202 (a)(1)	5.4.2	Total Effective Dose Equivalent Limit for Adult Workers
10 CFR 20.1201(a)(1)(ii)	10 CFR 835.202 (a)(2)	5.4.3	Any Individual Organ or Tissue Dose Limit for Adult Workers
10 CFR 20.1201(a)(2)(i)	10 CFR 835.202 (a)(3)	5.4.4	Annual Dose Limit to the Lens of the Eye for Adult Workers
10 CFR 20.1201(a)(2)(ii)	10 CFR 835.202 (a)(4)	5.4.5	Annual Dose Limit to the Skin of the Whole Body and to the Skin of the Extremities for Adult Workers
10 CFR 20.1208(a)	10 CFR 835.206 (a)	5.4.6	Dose Equivalent to an Embryo/Fetus
10 CFR 20.1301(a)(1)	DOE Order 458.1	5.4.7	Total Effective Dose Equivalent Limit for Individual Members of the Public
10 CFR 20.1301(a)(2)	10 CFR 835.602 10 CFR 835.603	5.4.8	Dose Limits for Individual Members of the Public in Unrestricted Areas
10 CFR 20.1301(b)	10 CFR 835.208	5.4.9	Dose Limits for Individual Members of the Public in Controlled Areas

The requirements in 10 CFR Part 20 that are relevant to WIR evaluations for DOE facilities (i.e., non-NRC licensees), are the dose limits for radiation protection of the public and the workers during disposal operations. However, the 10 CFR Part 20 requirements are not relevant in the context of general licensing, administrative, programmatic, or enforcement matters since DOE's WMA C will not be licensed by the NRC or an Agreement State.

The Draft WIR Evaluation for WMA C did not address 10 CFR 20.1206(e) because DOE does not plan any special exposures for closure operations at WMA C. The Draft WIR Evaluation also did not address 10 CFR 20.1207, because no minors will be working at WMA C who would receive an occupational dose. DOE explains that doses will be maintained ALARA and that the dose limits correspond to the dose limits in 10 CFR Part 835 and relevant DOE orders, which establish DOE regulatory and contractual requirements for DOE facilities and activities.

The DOE has a similar requirement as NRC regulation 10 CFR 20.1003 which defines ALARA. DOE regulation 10 CFR 835.2 defines ALARA as "… the approach to radiation protection to manage and control exposures (both individual and collective) to the work force and to the general public to as low as is reasonable…"

3.11.2 NRC Evaluation of Protection of Individuals During Operations (10 CFR Part 61.43)

DOE provided adequate information that individuals will be protected during operations. DOE provided a detailed crosswalk of the relevant DOE regulations to those provided in 10 CFR Part 20, which is referenced in the 10 CFR 61.43 performance objective. The NRC staff agrees that an equivalent level of protection is provided by the relevant DOE regulations as found in Part 20. In addition, DOE applies measures to ensure that exposures of individuals are maintained ALARA including: (1) a documented radiation protection program, (2) a documented safety analysis, (3) radiological design for protection of individuals, (4) regulatory and contractual enforcement mechanisms, and (5) access controls, training, dosimetry, and monitoring.

DOE limits effective dose equivalent from air emissions to the public at 0.1 mSv/yr (10 mrem/yr) in DOE Order 458.1 to comply with the EPA requirement in 40 CFR 61.9. The estimated dose per year from all operations at the Hanford Site including the WMA C closure operation 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 (DOE, 2018).

The NRC regulation at 10 CFR 20.1201(a)(i) concerning occupational dose limits for individual adults has a total effective dose equivalent limit equal to 5 rems (50 mSv). The DOE regulation in 10 CFR 835.202 has the same annual dose limit for the annual occupational dose to general employees. The occupational dose to adults during WMA C closure (the total effective dose per year) is to be controlled using the ALARA principles. Occupational doses to workers have been well below the annual limits specified in 10 CFR 20.1201(a)(1)(i) for all Hanford Site work activities, and the total effective dose to workers have been below the given limit (DOE, 2018). DOE’s quarterly radiological performance report based on the radiation protection program at the Hanford Site includes occupational radiation exposure results. For the period 2011 to 2015, the average dose for an exposed worker was 0.52 mSv/yr (52.2 mrem/yr) (WRPS-1603585, 2016).

The public will be located miles, or kilometers, away from the facilities during operations, and active security is maintained to prevent inadvertent access to the site. The NRC staff agrees with DOE that the risk to the public during operations should be minimal, and that the relevant regulatory limits can be achieved.

3.11.3 NRC Conclusion on 10 CFR Part 61.43 Compliance

The NRC staff concludes based on its review of the Draft WIR Evaluation for WMA C and referenced documents, that there is reasonable assurance that the 10 CFR 61.43 performance objective will be met during facility operation.

In the Draft WIR Evaluation for WMA C, DOE states that every reasonable effort will be made to maintain exposures to radiation as far below the dose limits as is practical, consistent with the purpose for which the activity is undertaken, and that 10 CFR 835.101(c) requires the contents of each radiation protection program to include formal plans and measures for applying the ALARA process to occupational exposure. In general, the NRC staff have reasonable expectation that WMA C closure will comply with the applicable dose limits and with the ALARA provisions because of the following: a documented tank operations contractor radiation protection program (HNF-5183, 2016), a documented safety analysis (RPP-13033, 2014), procedures for radiation protection design, regulatory and contractual enforcement mechanisms, and access controls, training, and dosimetry. In addition, in Section 5.4.11.6 of the Draft WIR Evaluation for WMA C, DOE discusses how the effectiveness of the radiation protection program is demonstrated by the occupational radiation exposure history for tank operations contractor.

3.12 Stability of the Disposal Site (10 CFR Part 61.44)

Stability, also referred to as *structural stability* in the regulations, is the capability of the wasteform, engineered features, disposal facility, and disposal site to maintain their intended properties to meet the 10 CFR 61.41 and 10 CFR 61.42 performance objectives of Subpart C. The Subpart C performance objective pertaining to "Stability of the Disposal Site after Closure" is 10 CFR 61.44 and states the following:

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

The site chosen for disposal, and the design used to isolate the waste, must be assessed for their ability to provide long-term stability. The conceptual designs should not require active maintenance or any systems that rely on active maintenance or active intervention, should disruptive processes or events cause site degradation. Disruptive processes or events that significantly degrade waste isolation by directly or indirectly affecting site, facility, or wasteform should be identified and evaluated. These may include processes and events that (1) affect the site such as erosion, flooding, or seismicity, (2) affect the facility such as degradation of an erosion barrier, or (3) affect the wasteform such as differential settling caused by voids in the waste.

3.12.1 Summary of DOE Results for Stability of the Disposal Site (10 CFR Part 61.44)

Appendix H in the WMA C PA (see Table H-1) identified two features of the system as safety functions relevant to the stability of the site, facility, or wasteform. These safety functions are *EB5 - Tank structure (structural)*, which includes the concrete dome and walls and is intended to provide structural support preventing subsidence of the closed facility, and *EB11 - Grout in tank (structural)* which is intended to provide structural support to prevent subsidence of the closed facility.

In the Draft WIR Evaluation for WMA C (DOE, 2018), DOE described design and closure activities that are intended to provide sufficient stability to meet the 10 CFR 61.44 performance objective. Prior to closure, the use/operation of WMA C waste tanks and ancillary structures will support long-term stability. During waste storage and retrieval operations, corrosion control and structural integrity programs are implemented including tank inspection programs and a tank leak detection system that monitors the tanks for structural integrity. Pre-closure subsurface engineered barriers such as the concrete domes/roofs and walls from tanks and ancillary structures and steel liners are also currently in place to provide structural stability. DOE estimated these features will remain intact for thousands of years and will continue to provide structural stability. Before closure, single-shelled tanks and ancillary structures also will be filled with grout to prevent subsidence and DOE predicted that the infill grout will remain undegraded for thousands of years and eliminate the need for active maintenance.

The pipelines will not be filled with grout; however, the pipelines will be covered by the planned closure barrier. The engineered surface cover, the concrete features of the tanks and ancillary structures, and the infill grout are passive structures such that no active systems (e.g., sumps, pumps, ventilation, instrumentation requiring maintenance) will be needed other than surveillance, monitoring, or minor custodial care. In addition, the waste tank structures and grout fill are intended to significantly limit water flow through the waste tanks, and the waste tanks and ancillary structures (including the pipelines) are expected to be covered with an engineered surface cover that further limits the water infiltration into the waste tanks and ancillary structures. Limiting water infiltration should reduce the rates of many degradation processes that might affect stability. DOE predicted that the current strategy for closing the WMA C site will provide long-term stability and stabilize the residual waste in the waste and ancillary structures, maintain structural integrity for hundreds to thousands of years following closure, prevent subsidence, minimize water intrusion, and minimize the likelihood of inadvertent intrusion into the system and disturbance of the stabilized residual wastes.

Closure of WMA C will occur in three major steps (DOE, 2018): 1) removal of waste from the single-shelled tanks and ancillary structures, excluding the pipelines, 2) disposal facility and wasteform stabilization, including waste isolation, and 3) engineered surface barrier placement. DOE has nearly completed step 1) while step 2) will involve filling the tanks with grout to stabilize and immobilize the residual waste and prevent further long-term degradation of the waste tanks. The specific formulation of the grout has not yet been established; however, DOE will tailor and finalize the specific formulation of the grout before its application in WMA C. Although the current grout formulation is not yet finalized, it is anticipated that the grout will only contain fine aggregates. The consistency of the grout will be like that of a mortar rather than concrete. The grout may contain cement, fly ash, and water. The free-flowing grout will harden and stabilize the structure in addition to immobilizing the residuals. Step 3) involves placement of an engineered surface barrier (i.e., cover) that is expected to provide protection from infiltration and intrusion. The specific design of the closure barrier has not been finalized, but it is likely to be based on the Modified RCRA Subtitle C barrier as discussed in Section 3.2.1.2.2 of the WMA C PA. The closure barrier is intended to be a passive system requiring no active maintenance.

Surface and subsurface processes and events exist that can cause site or facility stability to be impacted. Flooding is the most common surface process that requires investigation. The Draft WIR Evaluation for WMA C examines the Columbia River and other surface water features such as the Yakima River and the Cold Creek valley. The Columbia River is regulated by numerous dams. There are no dams within the Hanford Site such that the approximately 50 mi- (80 km)-long Hanford Reach does fluctuate after flowing past the Priest Rapids Dam upstream of the Hanford Site. The controlled flow of the Columbia River caused by the numerous dams results in a lower flood hazard risk; however, dam-failure scenarios are potential contributors that could result in high flood flows. In the Draft WIR Evaluation, DOE discusses studies that calculated the PMF for the Columbia River downstream of Priest Rapids Dam showing that the central portion of the Hanford Site would remain unaffected. Additional studies evaluated scenarios for potential failure of the Grand Coulee Dam. Discharge resulting from a 50 percent breach of the dam would inundate the 100 Area, the 300 Area, and nearly all of the city of Richland, however, would not impact the areas occupied by tank farm facilities in the Central Plateau. The Yakima River was determined not to be a flood hazard for the tank farm facilities, and the PMF discharge rate for the lower Cold Creek Valley indicated that flooding of WMA C would not be a credible scenario.

Subsurface processes that can impact site or facility stability include water table fluctuations, subsidence or differential settlement, seismicity, and volcanism. DOE indicates that water table fluctuations should not impact WMA C due to the 65 m (213 ft) thickness of the unsaturated zone. In the Draft WIR Evaluation for WMA C, DOE summarizes field and laboratory studies that have been completed at many of the tank farm sites and concludes that there are no areas of potential surface or subsurface subsidence, uplift, or collapse close to the WMA C. With the exception of the loose superficial wind-deposited silt and sand in some locations, the in-place soils are competent and form good foundations. DOE concludes that liquefaction of soils beneath the tank farms is not a credible hazard due to the depth of the groundwater (see Section 3.1.4.5 in the WMA C PA (DOE, 2016)).

A network of seismic stations in eastern Washington collects information on earthquakes of magnitude greater than 2.5 on the Richter scale. Regional tectonic characteristics of the Northwest includes numerous complex zones of active faults surrounding the flood basalts, which allow large-scale stresses to be released. Section 3.1.4.3 in the WMA C PA describes how the most frequent seismic occurrences at the Hanford Site are earthquake swarms that consist of multiple small energy events that fall within a small energy range. In the WMA C PA, DOE states that the largest single event earthquake recorded near the Hanford Site occurred in Milton-Freewater, Oregon, located ~80 km (50 mi) away in 1936 at a Richter magnitude of 5.75. A probabilistic seismic hazard analysis estimated that a 0.1 g horizontal acceleration would occur every 500 years and a 0.2 g acceleration would occur once every 2,500 years. In the Draft WIR Evaluation for WMA C, DOE discusses how the range of earthquake magnitudes suggested by data summarized for the Hanford Site and the associated range of motion is generally imperceptible compared to clearly felt movement that can result in minimal building damage.

Two types of volcanic hazards have affected the Hanford Site in the past 20 million years: continental flood basalt volcanism and volcanism associated with the Cascade Range. Activity associated with flood basalt volcanism occurred between 17 and 6 million years ago and is no longer active, suggesting that the tectonic processes that created the episode have ceased and is not considered to be a credible volcanic hazard (DOE, 2018). Section 3.1.4.4 in the WMA C PA discusses volcanism in the Cascade Range including three volcanoes that are 200 km (124 mi) or less from the Hanford Site and that have been active in the last 10,000 years (i.e., Mount Saint Helens, Mount Adams, and Mount Rainier). Due to the volcano's distances from the Hanford Site, the deposition of volcanic ash would be the only potential hazard. In addition, the conclusions in WHC-SD-GN-ER-30038 (2012) state the Hanford Site is sufficiently distant from the Cascade Range volcanoes such that hazards from lava flows, pyroclastic flows and surges, landslides, lahars, and ballistic projectiles are below a probability of concern.

3.12.2 NRC Evaluation of Stability of the Disposal Site (10 CFR Part 61.44)

In order to the effects on long-term stability of WMA C due seismic loading or potential settlement, and although fracturing of the vault concrete and tank grout is not expected to result in significant changes to structural stability, an analysis demonstrating long-term structural stability is needed. However, the formulation for the intended infill grout has not yet been established at the time of this report. Therefore, NRC staff recommends that for the final Waste Evaluation for WMA C, when the grout formula has been established and closure conditions are known (e.g., closure barrier design is selected), a structural stability assessment should be completed (Recommendation #49).

Subsidence is not expected due the presence of the infill grout. However, potential long-term subsidence could be possible if extensive leaching of the grout significantly reduces the mass of the grout and leads to an appreciable difference in volume between the original intact grout body and a severely leached and degraded grout body. Therefore, NRC staff is making a general technical recommendation that when the grout formulation is finalized, DOE should verify that significant leaching is not expected to occur (Recommendation #50).

Void spaces currently exist in pipelines and will continue to exist after the closure. DOE stated that the volume inside the pipelines is 100 m^3 (3600 ft^3), and that there would be additional volume associated with pipe liners and concrete ducts or encasements (NRC, 2018e). In addition, DOE indicated that subsidence has not been explicitly evaluated but that they did not believe the volume was sufficient to lead to subsidence. The NRC staff also does not expect the remaining void space to be sufficiently concentrated at one locality to cause subsidence for the majority of the pipelines because the void space in the encasements is small from a differential settlement standpoint. However, as discussed in Section 3.5, some of the encasements may have significant void spaces and their collapse may lead to sudden or gradual differential settlement or subsidence. Therefore, NRC staff recommends that for the final Waste Evaluation for WMA C, DOE assess where the estimated largest volumes of void space will be located at closure to preclude possible subsidence issues (Recommendation #51).

Although NRC staff made recommendations with regards to the completion of a formal structural stability assessment and determining the largest volumes of void space in the WMA C and their locations, NRC staff has reasonable assurance that the 10 CFR Part 61.44 performance objective will be met, and that intended grouting of the waste tanks and ancillary structures will minimize void space and prevent subsidence and differential settlement that could occur due to consolidation. The NRC staff found the design and closure activities to provide stability for the disposal site, disposal facility, and wasteform to be adequate, in addition to the analyses associated with siting considerations and disruptive processes and events such as flooding, liquefaction, seismicity, and volcanism to be complete. The safety functions EB5 - Tank structure (structural) and EB11 - Grout in tank (structural) can maintain sufficient structural support to prevent significant subsidence within the WMA C.

3.12.3 NRC Conclusion on 10 CFR Part 61.44 Compliance

Based on the information presented and reviewed in the Draft WIR Evaluation WMA C and the WMA C PA, NRC staff has reasonable assurance that the 10 CFR Part 61.44 performance objective will be met, and that grouting of the waste tanks and ancillary structures will minimize void space and prevent subsidence and differential settlement that could occur due to consolidation.

However, the final determination of physical stability will occur after DOE has selected a grout formulation engineered surface cover design. In addition, because the final cover design is not yet completed, there are uncertainties associated with how the cover will perform from an erosion protection standpoint. Recommendations #49 through #51 are discussed in this section and included in Table 5-1 of this report.

3.13 Model Support

Most modeling activities require validation for their projections to be of value. Whereas verification is determining that the equations were solved correctly, validation is determining that the correct equations were solved. These aspects lend themselves to model support. Arguably validation can be much more difficult than verification. Because PA models are used to project potential radiological impacts to hypothetical receptors well into the future, these models cannot be validated in the traditional sense. Instead, PA models are supported by multiple, diverse sources of information that may leave residual uncertainty. Performance assessment models are collections of other models (e.g., process models), where inputs and the effects (outputs) are integrated between the process models. Even though the overall PA model may not be validated in the traditional sense, some of the individual process models may be validated.

3.13.1 Summary of DOE Analyses of Model Support

DOE described their approach to model validation[10] in Section 6.4 of the WMA C PA (DOE, 2016). DOE reviewed scientific literature on model validation. DOE indicated that model validation is precluded in the traditional sense and, therefore, provides a summary of the documented theoretical or scientific basis for each of the PA model components. The summary addressed the suitability of the components for PA decision-making. The areas DOE summarized were[11]:

- Recharge rates
- Source-term release
- Vadose zone flow
- Saturated zone flow and transport
- Atmospheric release

DOE estimated recharge rates from a variety of studies conducted at the Hanford Site over the last 30 years. Recharge represents the downward flux of water below the evapotranspiration zone representing deep drainage. The two primary references DOE cited were PNNL-16688 and PNNL-14702. Recharge estimates are based on lysimeter records, tracer tests (chloride mass balance), and computer simulations to match field data. DOE indicated that recharge

[10] DOE used the terms “confidence”, “confidence building”, “confidence enhancement”, “validation”, and “validation process” interchangeably.
[11] Each of these areas is briefly summarized here. However, the reader is referred to the specific technical section of the WMA C PA for a more detailed discussion of each topic.

rates are available for natural and disturbed conditions, with and without different types of vegetation present. DOE was able to use results from large-scale, long-term lysimeter studies to define recharge values (and their uncertainties). DOE also used information from the Prototype Hanford Barrier (PNNL-18845). DOE developed uncertainty distributions to apply to recharge rates for different conditions and timeframes.

DOE's model support for source-term releases was based on characterization reports, empirical and process model-based information, measurements of effective diffusion coefficients, and literature values for distribution coefficients (Kd)s. DOE considered mineral-phase solubility-limited and matrix-degradation rate-limited processes for development of the release rates of contaminants from the waste. DOE used empirical and process model-based information to develop the waste form release models used in the PA. The effective diffusion coefficients DOE used were based on experiments (documented in PNNL-23841). Sediment-concrete half-cell experiments were conducted with different amounts of iron present and moisture in the sediment. Some of the monoliths were carbonated. The partitioning of radionuclides with grout and concrete was based on an assumed linear sorption isotherm and literature data (DOE, 2016).

DOE developed the geologic framework for flow and transport in the vadose zone from a borehole dataset (RPP-RPT-56356). The borehole dataset contains over 3,000 logged boreholes. The hydraulic properties assigned to the different hydrogeologic units were based on laboratory core-scale measurements of samples representative of H1, H2, H3 units and backfill material (RPP-CALC-60450). Each heterogeneous unit was replaced with an equivalent hydrologic media. DOE performed confidence building activities in the vadose zone representation by simulating observations from a large-scale water injection experiment, and by performing inverse calculations (Zhang and Khaleel, 2010). DOE indicated that the first and second moments (movement of moisture and spread of moisture) were in good agreement with the observed plume. DOE indicated that an extensive database of moisture content information has been developed from site characterization activities. DOE provided comparisons of observed vs. simulated moisture contents as support for the vadose zone modeling. DOE stated that the results show good agreement between simulated and observed, lending model support to the vadose zone modeling.

Appendix C of the WMA C PA describes the development of the model parameters for the saturated zone flow model. DOE used the Central Plateau Ground Water Model (CPGWM) as the basis for developing the model for WMA C. The CPGWM was based on over 30 years of groundwater experience at the Hanford Site (as documented in report PNL-10886, PNNL-13641, and PNNL-14398). DOE's comparison of measured and simulated water levels, or hydraulic heads, for wells upgradient and downgradient of WMA C showed good agreement for a timeframe of approximately 20 years.

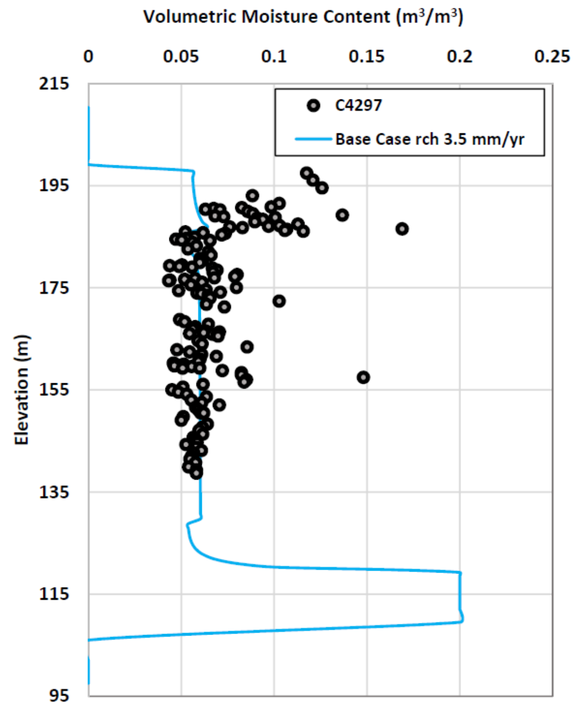


Figure 3-36 Observed vs. Simulated Moisture Contents (DOE, 2016)

DOE performed verification of air pathway modeling results by comparing the results from the PA with results generated using EPA CAP88-PC and with results provided in a Hanford National Environmental Policy Act characterization report (PNNL-6415). The comparison was provided in Appendix E of the WMA C PA. DOE indicated the comparison shows that the WMA C air pathway model is valid for its intended purpose.

DOE provided support for the different components of the PA modeling throughout the documentation. DOE provided sources of data, comparisons of data used to other sources of information, and comparisons to other modeling. One of the topics covered in the NRC staff's RAI was model support. In the RAI, the NRC staff indicated that DOE had not demonstrated that the simplified WMA C PA model included real-world features in a sufficient or conservative manner to support decision-making. The areas the NRC staff identified were in-leakage (advection) to existing tanks, radionuclide transport in the vadose zone associated with past leaks and spills, and contaminant transport in the saturated zone associated with past releases (NRC, 2019). DOE responded that it did not believe in-leakage was appropriate to model since they do not know the source of the water, modeling the transport of ^{60}Co past leaks was not possible or reasonable for a variety of technical reasons, and that saturated zone transport of ^{99}Tc was completed as part of analyses to model past leaks (RPP-RPT-59197, 2016) but that report was issued after the PA and was not part of the NRC staff's review. DOE summarized the evaluation in response to the RAI and stated that when uncertainties in groundwater fluxes were accounted for, results of the analyses showed arrival times and concentrations consistent with observed monitoring well data for ^{99}Tc (RPP-RPT-59197, 2016).

3.13.2 NRC Evaluation of Model Support

Performance assessments are calculations, but they are also models. Performance assessment models are used to project impacts into the distant future to make better decisions today. Model support is one of five general technical review procedures identified in NRC's guidance document for reviewing waste determinations (NUREG-1854). The goal of the review procedure 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.
- Perform simplified calculations to examine DOE outputs.
- Confirm that DOE has identified and implemented adequate procedures to construct and test its mathematical and numerical models.

The NRC staff's review of specific models/information (e.g., infiltration, waste release) are found in the respective sections of this TER. This section provides the NRC staff's review of DOE's overall strategy for model support of their PA model. Throughout the reports supporting the Draft WIR Evaluation for WMA C, DOE provided comparisons between outputs of components of the PA models and other sources of information (e.g., site characterization data, process-level modeling, laboratory tests, and field measurements). In many cases, the supporting information was used to develop the model rather than used to perform an independent confirmation of the model. Some components of the modeling had sufficient information to support the modeling, whereas in other areas, the information was more limited. One of the primary goals of the PA is to estimate when contaminants may reach members of the public and what concentrations of radionuclides the public may be exposed to. Many different models (i.e., submodels or process models) may be used to estimate intermediate outputs that are propagated to the final performance measures. Though doses to the public cannot be directly validated, model support can be provided for the intermediate outputs of the analyses.

DOE provided a model support section in the WMA C PA (Section 6.4). The modeling areas that DOE provided support for were recharge rates, source-term release, vadose zone flow, saturated zone flow and transport, and atmospheric release. DOE incorporated as much of the historically generated information into the PA as necessary to develop the models that were used. The NRC staff asked RAI 2-12 on the topic of model support for the PA model and identified specific areas where model comparisons with other sources of information would build confidence (NRC, 2019). The Hanford Site has been in operation for more than 70 years, with millions of documents generated over that time period. A large variety of scientific and technical studies have been completed and certain aspects of the site have been well-characterized. The NRC staff attempted to provide examples of information in the historical record that could be used to provide additional support for the PA modeling, since DOE's model support section was relatively brief and only included select aspects of the modeling. Though these topical areas are covered in the relevant sections of this report, some are discussed in more detail below to summarize the NRC staff's review and conclusions associated with DOE's strategy and implementation of model support for the PA.

DOE's base case release model was diffusion-only release from the tanks. As the NRC staff described in the RAI, there are numerous reports documenting rather large present day advective flows into the tanks (RPP-RPT-29191). Much of the flow may occur through spare inlet ports on the tanks that are poorly-sealed (some were sealed with wood). In response to the RAI, DOE indicated that because the source of the water is unknown, that water flow into the tanks cannot be used to support PA modeling. This approach is relying on uncertainty to do less, rather than investigating the source of the uncertainty in order to determine the importance of the phenomenon. All information, to the extent practical, should be used to support (or refute) the PA modeling. Particular attention and emphasis must be placed on information, even if uncertain, that may refute hypotheses about how the system works to avoid confirmation bias. The observation of significant amounts of water entering the tanks is a fact, not an assumption. If the source of the water is unknown to DOE, DOE should perform investigations to identify the source of the water and understand how and why it is occurring. DOE is assuming waste release from the tanks will be diffusion-only and dismissing data that suggest otherwise without investigation. DOE did investigate alternative cases where advection occurred through the tanks, but in DOE's "hybrid" approach of using a deterministic compliance case combined with probabilistic uncertainty analyses and sensitivity cases, the uncertainty in the flow processes associated with closed tanks is not reflected in the base case results. DOE's approach to evaluating uncertainties does not provide a clear impact of the combined effect of uncertainties on the compliance case (see Section 3.14).

DOE did not provide plans to seal the inlet ports in the tank (or other potential advective pathways) and have not yet designed or demonstrated that their grout formulation will have proper rheological properties and minimal shrinkage such that a shrinkage gap will not form between the grout and the tank walls upon curing of the grout. These assumptions will need to be verified by DOE in the future. Modeling should be performed to demonstrate that the PA model is capable of producing the real-world observation. DOE's tank model does not represent features such as inlet ports and assigns uniform properties across tank components. The NRC staff identified that because of the moisture characteristic curve properties DOE assigned, that even when DOE thought that they were sending infiltration through the tanks that most of it was being diverted in the numerical model (NRC, 2019). If DOE were to demonstrate that they could simulate in-leakage to the tanks in the present day, it would build confidence that advective flow was not being "missed" in the PA model simply because of inadequate discretization of the numerical model and the particular assignment of moisture characteristic curve parameters. When models cannot be validated in the traditional sense, models must be assumed to be incorrect until they can be demonstrated to be correct (or sufficient to bound the impacts).

In the RAI, the NRC staff suggested that DOE should consider the information available from past leaks and spills to provide support for vadose zone flow and transport (NRC, 2019). Though leaks and spills are not something that anyone wishes to occur, they can provide valuable information to understand complex geological and hydrogeological systems. In the

RAI, NRC provided examples of select reports that provided information that may be compared to flow and transport modeling. For example, Figure 4-5 of RPP-ENV-33418 Rev. 3 provided depth profiles of ^{60}Co, ^{137}Cs, and ^{154}Eu. In response to the RAI, DOE provided a list of problems associated with using the ^{60}Co data (DOE, 2019). DOE may have misinterpreted the breadth of NRC's comment. NRC was providing an example from a large set of examples of information collected with past leaks and spills that could be used to develop support for DOE's flow and transport modeling in the vadose zone. Some of the problems DOE identified with using ^{60}Co included an unknown origin of the leak which produced the observed contamination, ^{60}Co has a short half-live which makes interpretation difficult, ^{60}Co is not a risk driver, and vadose zone measurements are influenced by fingering and other kinds of local spreading. Though ^{60}Co is not a risk driver, other isotopes that may be important for future decisions associated with remediation of contaminated soils are also included in the reports. DOE did not address the data available from other isotopes in the response. NRC acknowledges there are challenges associated with using the past leak data. However, uncertain inferences associated with vadose zone flow and transport can still be quite useful. The flow and transport simulations cover thousands of years, and the leaks cover tens of years. If the PA model cannot estimate the transport of contaminants over tens of years it is not clear why confidence should be placed into estimates over much longer timeframes. One of the purposes of the PA model is to evaluate the impacts of uncertainties. The problems DOE identified could be accounted by including more uncertainties in the assessment.

DOE expressed caution with using past leak information because infiltration rates were different in the past, the timing, magnitude, and composition of the leaks were not known, liquid discharges perturbed the water table, and there was mixing of contamination from different sources. DOE's flow and transport models used in the PA should be robust enough such that the models can incorporate the uncertainties DOE identified. Because of the uncertainties, developing model support using data from past leaks and spills should be done probabilistically. DOE provided comparisons of simulated vs. observed moisture contents (replicated in Figure 3-36 above) to demonstrate that the vadose zone hydrology was adequately supported. The simulated moisture contents are not unique; different combinations of moisture characteristic curve parameters and infiltration rates can produce simulated moisture profiles that are in reasonable agreement with the observed data. Moisture flow rates (and directions) are even less constrained because they represent the derivative of moisture contents. Contaminant transport rates are even less constrained than moisture flow rates because they are influenced by geochemical processes. Selected distribution coefficients (Kd)s may be verified or constrained by simulating past leaks and spills. Without analyses of past leaks and spills to better constrain the vadose zone flow and transport simulation results, the compliance case results represent one possible outcome of a range of possible outcomes.

The NRC staff carefully examined DOE's performance assessment model. Staff performed independent verification of DOE's results. Staff modified DOE's inputs and calculations to examine additional cases, such as examining the impacts if the long-term infiltration rates were to be much higher than anticipated. Staff examined numerous uncertainties associated with model support. Because staff did not identify plausible cases where the 10 CFR Part 61.41 performance objective would be exceeded, staff concludes that the model support is sufficient for the Draft WIR Evaluation. The model support provided is sufficient for regulatory decision-making, however the model support provided does not provide sufficient basis for the deterministic base case results. The 10 CFR Part 61.41 performance objective is likely to be met with reasonable expectation but not by the margins projected by DOE. If similar models were used for future waste evaluations, the model support should be improved (Recommendation #52). A well-supported system model can oftentimes facilitate better understanding of the problem than a complex process model with lessor support.

Summary of Review

- The NRC staff reviewed DOE's model support. DOE's model support was sufficient for regulatory decision-making for WMA C. However, DOE's model support was not adequate to support the deterministic base case.

- There are no significant sources of uncertainty associated with this section.

- Recommendation #52 discussed in this section is also included in Table 5-1 of this report.

3.14 Uncertainty

Uncertainties may be pervasive in the types of technical analyses completed to support waste evaluations. To the extent practical, uncertainties should be identified, characterized, assessed, and if necessary mitigated.

3.14.1 Summary of DOE Uncertainty Analyses

DOE's guidance for completing the uncertainty and sensitivity analyses (DOE Guide 435.1-1) states that dose rates have uncertainties, and a discussion of the uncertainties should be included in expressing the outcomes of a PA. The guidance further states that an estimate of the magnitude of uncertainty is needed for the analysis that includes the calculation of the maximum impact of the disposal facility beyond the 1,000-year compliance period. DOE stated in the PA[12] documentation that the intent of the uncertainty and sensitivity analysis is to identify the assumptions and parameters that have the greatest impact on the projected doses and to evaluate the consequences of the uncertainties relative to the performance objectives (DOE, 2016). This is because exact or precise estimates of future impacts are not truly quantifiable, and even the sources of uncertainty remain unquantifiable because they must include elements of subjectivity.

3.14.1.1 Data Uncertainty

The first step in the uncertainty assessment process is to identify the potential sources of uncertainty. In Section 8.1 of the PA document DOE described the development of uncertainty ranges and probability distributions for parameters used in the PA modeling (DOE, 2016). DOE identified parameters that were expected to be uncertain.

[12] A PA is typically a quantitative evaluation using a model or collection of models to estimate uncertain impacts. The documentation of the analysis is commonly also referred to as the "performance assessment". The term is used interchangeably in this document and where necessary the usage is clarified.

The uncertain parameters DOE evaluated were:

- Recharge rates for disturbed and undisturbed areas for four different time periods (pre-operations [prior to 1945], operations [1945 to 2020], post-closure with intact cover [2020 to 2520], post-closure with degraded surface cover [> 2520].
- Residual inventory of different radionuclides in different components.
- Source term transport parameters (initial release fraction of ^{99}Tc, slower release rate of ^{99}Tc, solubility of uranium, effective diffusion coefficient of the base mat, sorption parameter (Kd) for different radionuclides on cementitious materials).
- Vadose zone hydraulic properties (K_s, α, n, θ_s, θ_r) for different media (H1, H2, H3, backfill).
- Sorption parameters (Kd) for the natural system.
- Darcy flux in the saturated zone.
- Macrodispersivity in the vadose and saturated zones.
- Gas-phase tortuosity.
- Wind speed.

3.14.1.2 Scenario, Conceptual Model, and Parameter Uncertainty

Scenario uncertainty is associated with incomplete knowledge in forecasting future states of the system being analyzed. If a disruptive event cannot be excluded on the bases of low probability or consequences, an alternative scenario would be required to assess the uncertainties associated with an alternative plausible future system state. For example, while the science of climate forecasting has improved substantially in recent decades, our knowledge of the climate and how it may evolve is still incomplete. DOE evaluated select cases of conceptual model uncertainty in their Draft WIR Evaluation. DOE evaluated model uncertainty using sensitivity cases and by developing alternative numerical models. DOE evaluated one source of stakeholder concern associated with lateral flow by implementing a different geological model and calculating new flow fields. Scenario and conceptual model uncertainty are discussed in detail in Section 3.2 in this report.

In addition to evaluating uncertainty in parameters, DOE evaluated uncertainty in the performance of safety functions using sensitivity analyses. This was done by defining specific cases to look at the impacts on the performance objective metrics by changing either the performance of a safety function. For example, if a barrier were to perform for a shorter period rather than the baseline performance. The safety functions evaluated were surface barrier flow, aquifer dilution, vadose zone flow and dispersion, inventory, flow through grout, waste release, and flow through tanks. DOE also included other sources of uncertainty by making conservative assumptions. Specific details on the technical information provided by DOE and NRC's review of that information can be found in the corresponding technical sections of this report.

3.14.1.3 Sensitivity and Uncertainty Analyses

DOE used different analyses to propagate uncertainties through the performance assessment calculations and evaluate the impact of those uncertainties. The primary methods were probabilistic uncertainty analyses and deterministic sensitivity analyses. Probabilistic uncertainty analysis was used to quantify and propagate the uncertainties in input parameters and assumptions. A goal of the uncertainty analysis DOE completed in the WMA C PA was to determine the significance or importance of a barrier to the resulting dose, and NRC staff will also refer to the uncertainty analysis as a barrier importance analysis in this report. DOE clearly described the parameters for which uncertainty was prescribed (e.g. probability distributions were assigned) or for which sensitivity cases were evaluated. The methods DOE used were appropriate and NRC was able to determine which parameters were evaluated in the analyses.

DOE's base case assessment was deterministic because flow and transport was represented by STOMP, a multi-physics flow and transport model (PNNL-12030). The STOMP model was not fully integrated with the other process models in an automated process that would be necessary to complete the large number of realizations (different iterations of the calculations) for a probabilistic evaluation. To perform the uncertainty assessment, DOE developed an abstracted PA model using GoldSim© (GoldSim Technology Group, 2009). Abstraction is the process of simplifying a model such that the essential response of the model to changes in inputs is retained but the abstraction is simpler and easier to integrate and execute in a more computationally efficient manner than the detailed model. For instance, a lookup table or response surface could be an abstraction. NRC concurs with DOE's use of abstractions in the probabilistic system model. A well-supported system model can oftentimes facilitate better understanding of the problem than a complex process model with lessor support.

DOE utilized the Monte Carlo method for the uncertainty assessment. This method has been widely used for evaluation of radioactive waste disposal. In this method, discrete sets of input parameter values are selected at random from probability distribution functions. Then the model is run for each set of sampled parameter values, and a probability distribution function of model output is constructed. That distribution represents the uncertainty in model output associated with uncertain input parameters. DOE used a type of sampling called Latin Hypercube Sampling (LHS) to sample the probability distributions. The LHS approach ensures coverage of all ranges of the distributions and can lead to faster convergence of the results.
The Monte Carlo method used by DOE had the following steps:

- Select model input parameters
- Assign probability distributions to input parameters to quantify uncertainty
- Generate many sample sets (realizations) through sampling of probability distribution
- Propagate the uncertainty (via realizations) through the analysis
- Determine parametric and nonparametric estimates of the reliability in the model output once an appropriate sample size is reached that ensures stable estimates of the output distribution.

DOE implemented a Monte Carlo approach in the WMA C PA using GoldSim with stochastic variables that represent the range of uncertain parameters in the WMA C model. DOE considered guidance provided in Technical Report TR-02-11, "Assigning probability distributions to input parameters of performance assessment models" and EPA guidance to define probability distributions for uncertain parameters (Technical Report TR-02-11, 2002; EPA/630/R-97/001). DOE used a maximum entropy approach while attempting to avoid risk dilution that can occur from the use of overly broad probability distributions. DOE computed a variety of metrics to assess the significance of uncertainties including correlation coefficients based on ranks, standardized regression coefficients based on ranks, partial correlation coefficients based on ranks, and importance measures based on ranks. The analyses were performed for different time periods (e.g. at the end of the compliance period [1,000 years] and at the time of peak dose [3,400 years]).

3.14.2 NRC Evaluation of Uncertainty

This section provides the NRC staff review of DOE's approach to uncertainty. DOE's stated intent of uncertainty and sensitivity analysis was to identify the assumptions and parameters that have the greatest impact on the projected doses and to evaluate the consequences of the uncertainties relative to the performance objectives. Specific details on the technical information provided by DOE to define the uncertainties in parameters and NRC's review of that information is found in the corresponding technical sections of this report. NRC evaluated the uncertainty analyses methods DOE utilized and the results of the analyses.

DOE's base case model is deterministic therefore uncertainties were not included in it. To evaluate the impact of uncertainties, DOE developed a probabilistic system model with a simplified abstraction of flow and transport. DOE included adjustments to some inputs to the system model to provide better agreement between the system model and the deterministic model using STOMP. It is appropriate to use abstracted models in the performance assessment and DOE demonstrated that the abstracted model was a reasonable representation of the deterministic base case model that relied on more complex process modeling.

Figure 3-37 is the groundwater pathway dose results from the probabilistic analysis. The median dose in 1,000 years is less than $1x10^{-5}$ mSv (0.001 mrem) and the peak median dose is approximately $1x10^{-3}$ mSv (0.1 mrem) (DOE, 2016). The range of doses is approximately 2-3 orders of magnitude, though the range can be larger when doses are increasing due to shifting in time of when the doses begin increasing. The largest dose at any time in any realization was 0.025 mSv (2.5 mrem), well below the performance standard of 0.25 mSv (25 mrem). DOE's probabilistic dose results are consistent with NRC staff's experience from the review of complex performance assessments. There is reasonable agreement between the deterministic base case and the dose results from the probabilistic system model.

DOE evaluated uncertainties in the performance assessment through multiple methods and communicated the impacts of those uncertainties on the results. The use of multiple methods to characterize and assess uncertainty, or to mitigate the impacts of uncertainties, was appropriate. Staff concurs that DOE considered uncertainty when developing the performance assessment.

NRC reviewed the probabilistic analyses and finds that the method used (Monte Carlo with LHS sampling) was appropriate. The use of multiple metrics (e.g., importance measures based on ranks) was appropriate and can help give a more robust understanding of which uncertainties are significant than use of a single metric alone. The sensitivity measures DOE used were metrics that are commonly used in the performance assessment field. In particular, the NRC staff previously evaluated different sensitivity measures and found the importance measure to be reasonably robust for different types of data/models (Esh, 2016).

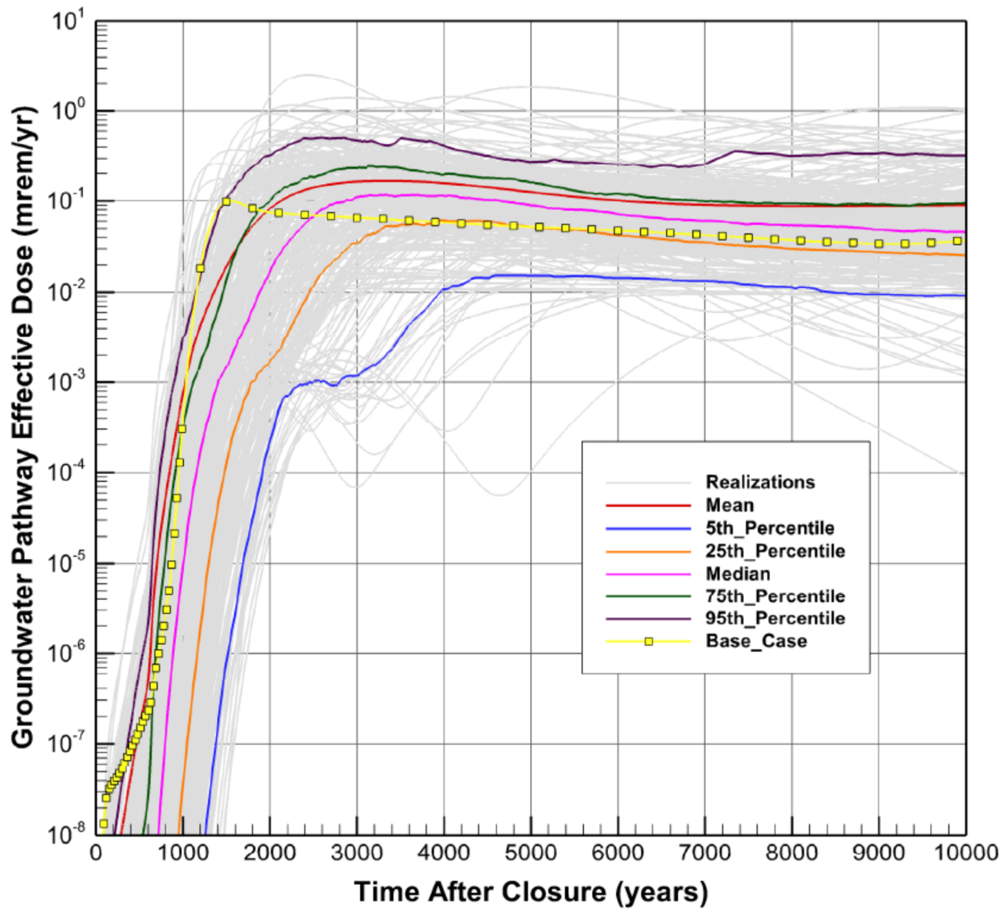


Figure 3-37 Groundwater Pathway Dose Results from the Probabilistic Analysis
(DOE, 2016)

DOE identified parameters that were expected to be uncertain. DOE used engineering judgment to decide which parameters to include as uncertain. The parameters they identified were associated with recharge rates, inventory of waste, source term transport parameters, vadose zone parameters, sorption parameters, Darcy Flux in the saturated zone, macrodispersivity, and gaseous transport. The total number of uncertain parameters that DOE considered was 130. However, that number is a bit misleading because three areas (inventory, sand Kds, and grout Kds) each required a vector of 38 inputs. In addition, recharge required three parameter distributions for one input. The total number of uncertain parameters was 130 but those parameters effectively represent 19 inputs, and 3 of those inputs were for the air pathway. NRC staff reviewed the set of uncertain parameters and found them to be consistent with staff experience and the results from other performance assessments.

The parameters that DOE identified as important to the groundwater dose in the 1,000-year timeframe were the Tc Kd for sand (Kd_Sand_Uncert[Tc]), the saturated zone dispersivity (SZ_Dispersivity), and the vadose zone dispersivity (VZ_H2_Dispersivity_Uncert). Staff agrees with the DOE's conclusions and would note that the most important parameter was the Kd for sand because it strongly affects the time of the Tc dose (which then translates into a magnitude impact). For the time of peak dose (3,400 years after closure), DOE identified 6 parameters as important (vadose zone flow field selector [Hyd_Prop_Uncert], Darcy flux multiplier in the saturated zone [Darcy_Flux_Mult_SZ], long-term recharge after barrier failure [Recharge_Late_PC_Uncert], saturated zone dispersivity [SZ_Dispersivity], Tc Kd for sand [Kd_Sand_Uncert[Tc]], ^{99}Tc inventory uncertainty [Tc99_Inv_Mult]). The identified parameters are all sensible and consistent with ^{99}Tc being the risk driver and being a relatively conservative contaminant from a transport standpoint. Some of the measures of sensitivity were small (e.g., importance measures less than 0.2) and may indicate a weak sensitivity or even a spurious result. The use of multiple measures helps ensure the robustness of the results and should continue to be a standard practice. NRC staff recognizes that there is a difference between relative sensitivity and absolute sensitivity. A large increase in a small dose is not significant from a regulatory decision-making perspective.

The NRC staff also examined the PA model to determine what inputs were not evaluated as being uncertain but possibly could have been. Some of these inputs can be highly variable. Some of these uncertainties were evaluated with sensitivity cases[13]. However, others were not evaluated (e.g., uncertainty associated with recharge rates, biosphere parameters such as soil to plant transfer factors, mass loading values, locally-consumed plant fraction, land-use impacts on hydrology). Some of these parameters are not well-known for the Hanford Site or have shown high variability. It is recommended for future waste evaluations that DOE include more parameters in an initial uncertainty assessment then eliminate parameters that are not found to be significant for a final uncertainty assessment (Recommendation #53). Though most of these inputs were obtained from values compiled in a variety of literature sources that does not mean the underlying values are constant and should be fixed as constant in the performance assessment. The goal of the system model is to evaluate the significance of uncertainties. If inputs are fixed as constants the significance cannot be readily reflected in the results. The areas where constants were assigned but could be reevaluated include:

- Biosphere parameters (soil to plant transfer factors, consumption rates and fractions, airborne mass loading)
- Material properties (density, porosity, saturations [engineered])
- Intruder parameters (drill diameter, waste thickness, cuttings management)
- Atmospheric transport parameters (Henry's Law, wind averaging period)

DOE indicated that they used a maximum entropy approach to assigning probability distributions but used caution when assigning the ranges to avoid "risk dilution." Risk dilution is when the magnitude of the peak dose in a probabilistic simulation is reduced because of the arbitrary assignment of overly broad probability distributions. In Section 8 of the PA, DOE presented the questions they considered when prescribing probability distributions (DOE, 2016). Staff review of the specific ranges and distributions assigned for uncertain parameters is found in the respective section of this report. Overall, some of the parameter distributions DOE assigned were too narrow and truncated to observations. When a uniform or triangular distribution is applied, the analyst is asserting that they are 100% confident that values outside of the endpoints of the distribution cannot occur. With most data, unless there are physical

[13] As discussed in this report, NRC staff does not support the use of sensitivity cases when they are the primary method to examine uncertainties where the effects propagate through many system components.

bounds, this situation rarely occurs. Risk dilution is a concern, especially for assigning probability distributions to parameters that influence the timing of doses. However, risk dilution can be properly accounted for by first using broad distributions that account for what may be limited observations. If the parameter is found to be significant, additional information can be collected to reduce the range of the probability distribution. If additional data cannot be collected, then the range of the probability distribution of the parameter can be selected to represent the conservative, or pessimistic, portion of the distribution in terms of overall system performance. Narrowing the distribution based solely on expert opinion can result in a non-representative distribution as it has been narrowed due to lack of knowledge (i.e., less observations results in a narrower range). Selected examples include the following (however, this issue was pervasive throughout the assignment of probability distributions):

- The initial release fraction of ^{99}Tc was given a uniform distribution with a range of 4.5% to 15%. The reference where the information was derived described limited experimental observations for tanks C-103, C-202, and C-203 (PNNL-20616). If additional experiments were completed for the same tanks or especially for other tanks it is extremely unlikely that the new data would all fall within the range of the prescribed uniform distribution.
- The van Genuchten “alpha” parameter distribution was developed by fitting a log-normal distribution to observed data. The data was fit with a log-normal distribution then truncated at the minimum and maximum observed values. Unless there are physical constraints, the distribution should not be truncated.
- The Kd value for uranium in sand was assigned a triangular distribution with a minimum of 0.2 ml/g and a maximum of 2.0 ml/g based on RPP-RPT-46088. The referenced report provides a distribution based on engineered judgment (and references other reports). PNNL-13895 provides many derived uranium Kds that are outside the range of the assigned distribution (some are negative, suggesting zero as a lower bound).

DOE made use of numerous site-specific measurements in developing their performance assessment. NRC commends DOE for collecting site-specific information and that work should be continued. Future performance assessments should do a better job accounting for the uncertainties associated with the representativeness of observations (e.g., does the rate of uranium release from Tank C-202 represent the rate of uranium release from Tank C-110) and the limited observations when assigning probability distributions. The initial uncertainty assessment should ensure that the tails of the distributions are appropriately broad (Recommendation #54).

DOE evaluated uncertainty in the performance of safety functions using sensitivity analyses. This was done by defining specific cases to look at the impacts on the performance objective metrics by changing the performance of a safety function. The sensitivity analyses looked at a combination of parameter, model, and scenario uncertainty. Table 3-18 provides a listing of the cases and a brief description (see Table 8-15 in DOE, 2016).

The surface barrier flow safety function showed a potential large impact on the timing of peak dose but only a moderate impact on the magnitude of peak dose (less than +/- a factor of 2). For the aquifer dilution safety function, the 5^{th} percentile values produced peak doses that were 2.6 times larger and the 95^{th} percentile value produced peak doses that were 1.6 times smaller than the base case. Though this is a significant impact on a relative basis, the doses are small on an absolute basis. The cases to examine the vadose zone flow and dispersion safety function showed a limited impact on the timing and magnitude of maximum concentrations in a down-gradient well. The largest increase in the base case dose magnitude was 60 percent. A more significant change was observed when using the upper bound inventory (case inv2). The maximum groundwater concentration was a factor of 4.5 larger than the base case. The grout flow sensitivity cases resulted in only moderate changes from the base case (similar results up to an increase of less than 50%). The residual chemistry safety function case produced very little change to the results for ^{99}Tc from the base case. Likewise, the tank flow safety function cases produced at most a moderate change in the base case results.

NRC staff reviewed the safety function cases and found the results to be reasonable and consistent with the documentation describing the performance assessment model. The responses of the modeled results were consistent with changes to the input parameters. The type of analyses DOE performed is useful to develop understanding of the modeled system and to test the resilience of the results to uncertainties or changes to the system. The results show that the system is resilient to failure or underperformance of any one safety function.

A challenge the staff encountered when reviewing the results of the safety function analyses was placing the results in the proper context. As noted previously, DOE used the safety function analyses to evaluate different types of uncertainties and included a mixture of uncertainties that were plausible with those that would generally be regarded as implausible. In addition, NRC staff asked questions during the review process about various technical issues and DOE indicated those issues were addressed in the sensitivity cases (see conference call summaries listed in the reference section). The sensitivity cases addressed various levels of underperformance of one area of the PA (e.g., recharge) thus they cannot address the impact of uncertainties in numerous different components of the PA concurrently. A probabilistic analysis or a full or partial factorial sensitivity case analysis is necessary to look at the combined impact of uncertainties. NRC provided a number of uncertainties in the RAI that either were not addressed or were not adequately addressed in the PA (NRC, 2019). These uncertainties included:

- Long-term infiltration rates (e.g., sand dune formation, plant evolution)
- Performance of the yet to be designed engineered cover
- Erosion performance of the engineered cover
- In-leakage to systems and advective release
- Lateral diffusion from the source term
- Grout shrinkage of the yet to be designed grout
- Organics impacts on waste release and retention of radionuclides
- Sulfate impacts on grout
- Presence of chelating agents in the waste
- Corrosion of penetrating steel
- Integrity of the basemat concrete
- Seismic impacts on performance
- The representativeness of tank sampling
- The uncertainty in the inventory modeling systems
- The uncertainty in the saturated zone hydraulic conductivity

Table 3-18 Sensitivity Cases to Evaluate Safety Functions

Case ID	Safety Function	Description
inf01	Surface Barrier Flow	Surface barrier functions indefinitely
inf02	Surface Barrier Flow	Long-term infiltration higher (5.2 mm/yr)
inf03	Surface Barrier Flow	Long-term infiltration = 100 mm/yr
unc01	Surface Barrier Flow	Long-term infiltration = 0.5 mm/yr
unc02	Surface Barrier Flow	Infiltration = 100 mm/yr for all areas all times
gwp01	Aquifer Dilution	Aquifer flow parameters at 5^{th} percentile
gwp03	Aquifer Dilution	Aquifer flow parameters at 95^{th} percentile
vzp01	Vadose Zone Flow and Dispersion	Vadose zone hydraulic properties at 5^{th} percentile
vzp02	Vadose Zone Flow and Dispersion	Vadose zone hydraulic properties at 50^{th} percentile
vzp03	Vadose Zone Flow and Dispersion	Vadose zone hydraulic properties at 95^{th} percentile
vzp04	Vadose Zone Flow and Dispersion	Alternative geologic model II
vzp05	Vadose Zone Flow and Dispersion	Clastic dikes
inv1	Inventory	Inventory from the 2012 EIS
inv2	Inventory	Upper bound inventory
grt1	Grout Flow	Grout degrades at 5,000 years to sand
grt2	Grout Flow	Grout degrades at 1,000 years to sand
grt3	Grout Flow	Grout degrades at 500 years to sand
grt4	Grout Flow	Grout has properties of sand at 0 years
rls1	Residual Chemistry	All waste available for immediate release
dif1	Tank Flow	Effective diffusion coefficient of 1×10^{-7} cm^2/s
dif2	Tank Flow	Diffusion coefficient changes from 1×10^{-14} to 3×10^{-8} cm^2/s over 500 years after closure
dif4	Tank Flow	Tank intact for 5,000 years, followed by immediate release

In response, DOE performed an additional analysis to look at the combined impact of the uncertainties raised by the NRC (DOE, 2019). DOE emphasized that the analysis did not represent a credible representation of system behavior and that the analysis was not valuable for decision making. They indicated that the analysis provided an overly conservative estimate of facility performance and may lead to an unduly pessimistic idea of system performance. The assumptions DOE made were:

- No surface barrier for the duration of the analysis, with the infiltration rate set to its upper bound of 5.2 mm/year;
- The grout and base mat behave as gravel, so that releases are by advection from the beginning of the analysis;
- The residual waste is available for leaching at the beginning of the analysis; and
- Minimum dilution occurs in the aquifer, with the groundwater flux set at 5^{th} percentile value.

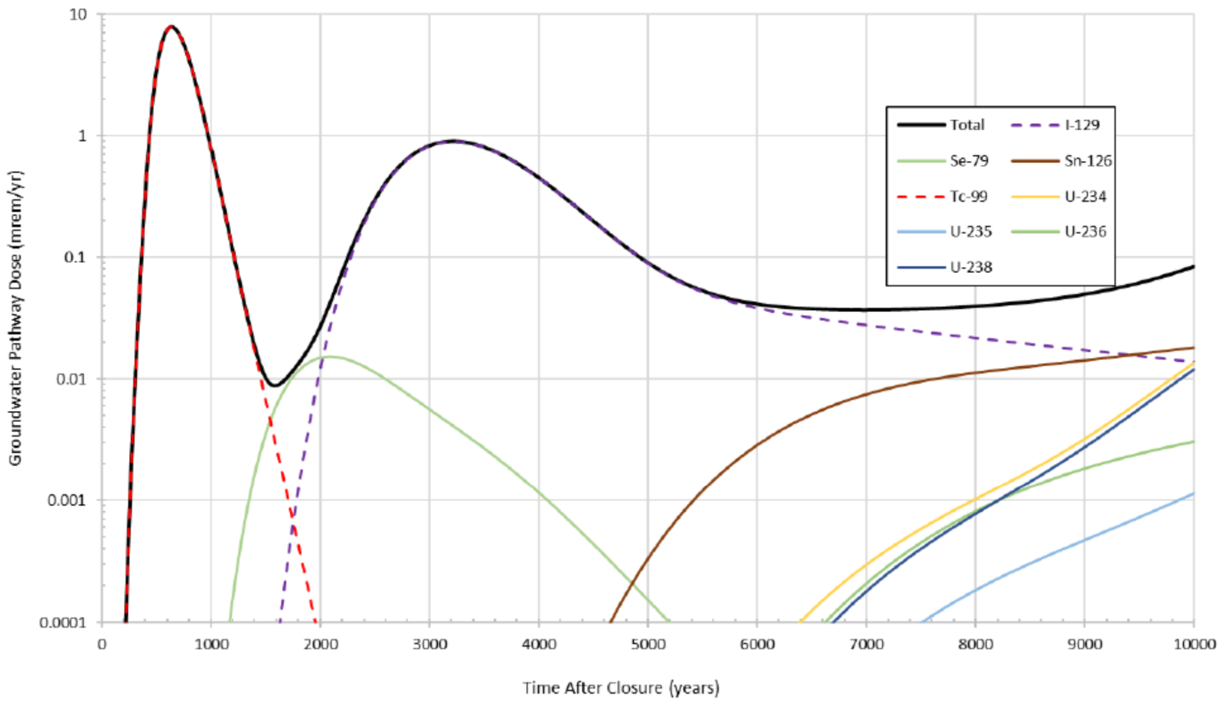


Figure 3-38 Combined Failure of Multiple Safety Functions

The results of the analysis are shown in Figure 3-38. The peak dose within the compliance period has increased to approximately 0.08 mSv/yr (8 mrem/yr) from $4x10^{-6}$ mSv/yr ($4x10^{-4}$ mrem/yr) (a factor of 20,000 increase). NRC agrees that some aspects of the evaluation are pessimistic, but the analysis is useful to support regulatory decision-making. In a system model with numerous uncertainties that were not evaluated in the probabilistic system model analysis, 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 to evaluate the global impact of uncertainties. This approach should not be used in future waste evaluations (Recommendation #55). NRC's conclusions about the appropriateness of uncertain evaluations are based on the system model and the global uncertainty evaluation case evaluated in response to NRC's RAI. The safety function analyses are useful to evaluate and understand the role individual safety functions may have, but safety function analysis is most useful if it is done with a full or partial factorial analysis, as was done previously by the NRC (CNWRA, 2011).

The analysis DOE performed should bound the uncertainties listed by the NRC, but it did not include uncertainties in the waste inventory. As discussed in Section 3.5, uncertainties in waste inventory may have been underestimated by DOE. Inclusion of inventory uncertainty may appear to result in the peak compliance period dose being over the performance limit (0.25 mSv/yr [25 mrem/yr]). The peak dose from the case was 0.08 mSv/yr (8 mrem/yr) and the uncertainty in inventory could be more than a factor of 3 increase or decrease). However, the middle two bullets (essentially the engineered barriers are not present) are likely to be pessimistic by at least a factor of 10. NRC does not agree that the assumptions about recharge and aquifer dilution are overly pessimistic. The peak dose is almost completely from ^{99}Tc. The timing of when recharge may increase (when the engineered cover may degrade) is not significant with respect to the magnitude of the peak dose. The long-term recharge magnitude is more significant, and the value used in this analysis is only 1.7 mm/yr (0.07 in/yr) more than the base case value. This increase may partially account for complex scenario uncertainties such as sand dune formation, plant succession, and range fires but the value is not clearly conservative or pessimistic given the scenario uncertainties. Likewise, the 5th percentile aquifer dilution is within the range of what the value could be based on currently available information. DOE has observed that operations at the Hanford Site created a significant groundwater mound at Hanford, and as the mound relaxes the flow directions can change, and even reverse. Under this scenario the effective dilution can be lower. Scenario uncertainties associated with future land use and river stages are plausible. For future waste evaluations, NRC recommends that 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 (Recommendation #56).

Summary of Review

- The NRC staff reviewed DOE's evaluation of uncertainty. The staff considered DOE's system model, safety function analyses, and combined uncertainty case developed in response to NRC's RAI. Staff performed independent calculations with DOE's system model. The system model was clearly-described and used appropriate methods to probabilistically simulate the impacts of uncertainties. The uncertainty measures used were appropriate and identified uncertain parameters. The impact of parameter uncertainty was evaluated globally. NRC relied on DOE's combined sensitivity case complete in response to NRC's RAI in order to determine that DOE properly evaluated the global impact of parameter and model uncertainties.

- Uncertainties included using parameter ranges that were truncated or too narrow, including too few parameters in the evaluation, not including plausible conceptual and model uncertainties in the system model, and performing one-at-a-time safety function evaluations.

- Recommendations #53 through #56 discussed in this section are also included in Table 5-1 of this report.

3.15 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.

3.15.1 Summary of DOE Quality Assurance Procedures

The sections that follow summarize DOE's QA procedures that were applied to development of the Draft WIR Evaluation and supporting analyses. Quality assurance applies to numerical model development as well as the collection and interpretation of data.

3.15.1.1 Numerical Model Development

Model development and application for the WMA C analyses were performed by DOE under a project plan that implemented the following:

- 10 CFR Part 830, Subpart A - Nuclear Safety Management Quality Assurance Requirements.
- DOE Order 414.1D (DOE, 2011b), Quality Assurance.
- State and federal environmental regulations.
- EPA guidance EPA/240/R-02/007, Guidance for Quality Assurance Project Plans for Modeling.
- EPA requirements EPA/240/B-01/003, EPA Requirements for Quality Assurance Project Plans.

The development, application, and preservation of environmental models used to support regulatory decision-making and analysis was conducted under a general project plan. The plan provided for modeling to be performed in a quality assurance framework for the full lifecycle, including control of models, implementation of software, application, and QA of supporting information. DOE provided Figure 3-39 to show the lifecycle quality applied to the environmental models (figure replicated from page 11-5 of DOE, 2016). DOE's project plan required training for modelers, that software quality assurance is applied (such as configuration management and testing), that models are documented, controlled, and preserved, and that full checking and model review is performed. The preparers of the modeling, analyses, and reports were identified in the WMA C PA (DOE, 2016). The preparers had advanced degrees in a variety of different disciplines and many years of experience in the field.

A variety of software packages were used for the analyses. The two primary packages used for source release and environmental flow and transport 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 uses a tracking system, the Hanford Information Systems Inventory (HISI), to manage software. For safety software, the HISI entry is used to record approval for use, authorized users, and to log instances of the software's usage. STOMP© was developed by PNNL which maintains a program to test the code to meet ASME NQA-1-2008 "Quality Assurance Requirements for Nuclear Facility Applications" as well as the DOE requirements found in DOE Order 414.1D. No errors were encountered with the use of STOMP© for the WMA C PA. DOE determined that GoldSim© Pro met the requirements for ASME NQA-1-2008 and DOE Order 414.1D. The responsibilities for using GoldSim© Pro included but were not limited to modeler training, source code installation and testing, software validation and verification, and reporting and documenting errors (none were encountered). Additional software (Tank Waste Information Network System (TWINS), Hanford Tank Waste Operations Simulator (HTWOS), Video Camera Computer-Aided Design Modeling System (CCMS), Hanford Defined Waste Model (HDW), Oak Ridge Isotope Generation and Depletion Code 2 (ORIGEN2)) was used in the analyses for the Draft WIR Evaluation for WMA C.

DOE identified the four basic model components necessary to provide traceable, reproducible models are the basis for model inputs (including data packages), the models, the application of the models, and the implementing software. Because models are more than software, DOE developed and implemented the Environmental Model Management Archive (EMMA) to maintain traceability and reproducibility. EMMA is a file system with synchronization to control and manage the model components. DOE makes use of Environmental Model Calculation Files (EMCFs). These EMCFs are prepared, documented, reviewed and approved per TFC-ESHQ-ENV_FS-C-05, "Preparation and Issuance of Model Package Reports and Environmental Model Calculation Files", an implementing procedure. Time limits are not provided for reviewers. Each package is assigned a checker, a senior reviewer, and a responsible manager. Review forms are used to record the details of the review and to track the identification and resolution of errors.

A numerical model was used to estimate future performance of the real-world system. DOE provided a detailed description of the system being modeled, such as the geology and engineered systems. Engineered drawings and photos of the tanks and ancillary equipment were provided.

Figure 3-40 provides an overview of the model development information provided by DOE (DOE, 2016). Information about the real-world systems were used to develop conceptual models that in turn were used to produce numerical models.

As part of NRC's review of DOE's Draft WIR Evaluation and supporting information, NRC identified discrepancies or inconsistencies in documentation, inputs, or other aspects of the calculations (NRC, 2019). DOE provided a response to RAI 2-1 in ORP-63747 Rev. 2 (DOE, 2019). For the most part, DOE indicated that they believed most of the items NRC identified were modeling assumptions and not quality assurance issues.

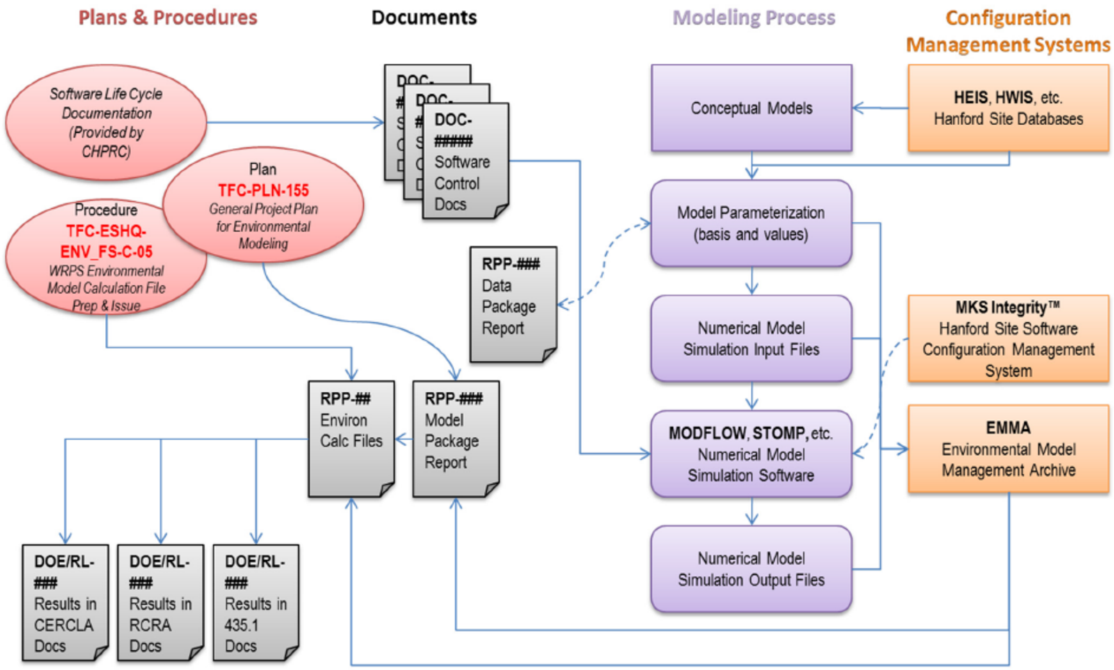
3.15.1.2 Data Validity

Data validity entails two primary aspects – that the data used is transparent and traceable and that the data used in the analysis is of acceptable quality. The DOE PA document (and supporting documents) provided the inputs to the calculations. DOE generally listed the source of the data next to the data and provided the list of references used. If data was provided in tables and if short-hand references were not provided in the tables (e.g., PNNL-16663), then footnotes to the tables were typically used.

The second part of the QA requirements for data validity is that the data has been qualified for use in the analyses. According to NRC guidance, data may be qualified by comparison to other data, or by documentation and evaluation such as by independent peer review (NUREG-1854). DOE uses quality assurance procedures when collecting or measuring data at their sites. The performance assessment relied on a large amount of "external" data, or data that was not developed or originated by DOE. Most of the external data came from widely cited references from other scientific or governmental organizations. DOE did not indicate how they determined the external data was of acceptable quality to use in the performance assessment calculations.

3.15.2 NRC Evaluation of Quality Assurance

NRC considered the guidance provided in chapter 8 of NUREG-1854 when reviewing DOE's quality assurance program and implementation. The main areas of review are data validity and software selection, development, and implementation.



435.1 = DOE O 435.1, 2001, Radioactive Waste Management, U.S. Department of Energy, Washington, D.C.
CERCLA = Comprehensive Environmental Response, Compensation, and Liability Act of 1980, 42 USC 9622, et seq.
CHPRC = CH2M HILL Plateau Remediation Company
HEIS = Hanford Environmental Information System
HWIS = Hanford Well Information System
RCRA = Resource Conservation and Recovery Act of 1976, 42 USC 6901, et seq.

References:

TFC-ESHQ-ENV_FS-C-05, "WRPS Environmental Model Calculation Preparation and Issuance," Washington River Protection Solutions LLC (WRPS) Tank Farms, Richland, Washington.

TFC-PLN-155, "General Project Plan for Environment Modeling," Washington River Protection Solutions LLC (WRPS) Tank Farms, Richland, Washington.

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MODFLOW software has been developed and distributed by the U.S. Geological Survey, Washington, D.C.

Battelle Memorial Institute (Battelle) retains copyright on all versions, revisions, and operational modes of the Subsurface Transport Over Multiple Phases (STOMP©) software simulator, as permitted by the U.S. Department of Energy. STOMP© is used here under a limited government use license.

Figure 3-39 Lifecycle Quality for Environmental Models (DOE, 2016)

The data validity review procedures are to ensure data are transparent and traceable, data have been obtained or qualified under a QA program or are otherwise documented and validated, and that data reduction processes have been documented. The software selection, development, and validation review procedures are to ensure that software is planned, controlled, developed, tested, controlled, and documented, configuration management is used, software is maintained, and defects or errors are properly managed. In addition, the linkages to the conceptual model and inputs are clearly described. Model limitations should be identified, and software management should ensure that software is not used outside of its validated range. NRC verified that DOE's model and software development process was managed under a project plan that implemented QA procedures. NRC reviewed the high-level documents (e.g., the Draft WIR Evaluation, PA document), supporting documents, software inputs and outputs, GoldSim model files, and other supporting calculations and analyses (e.g., Excel files). The NRC review was thorough, however the amount of information supporting the Draft WIR Evaluation was very large. The high-level documents were reviewed completely; the underlying documents were reviewed as needed. The staff used vertical and horizontal slices to select information to review.

NRC verified that the DOE staff who performed the analyses were qualified and trained in the QA requirements. DOE developed and implemented a management system to maintain traceability and reproducibility of the environmental modeling. A key component of QA for complex modeling such as a performance assessment is independent review and checking. NRC requested and reviewed EMCFs to examine how products are evaluated before they are released. DOE provided two examples: RPP-CALC-60448, Rev. 0, and RPP-CALC-60451, Rev. 0. These documents provide records of what software was used, who performed the review, and analyses that were performed. The EMCFs also provide a checklist (Form A-6006-716) that reviewers initial if inputs are documented, values are checked against parameter sources, and if the input in the EMCF matches the model input files. In RPP-CALC-60448, Rev. 0, the author examined discretization and numerical dispersion of the STOMP model. This was a good example of an independent review of select aspects of the modeling. This document had sufficient detail associated with configuration management and control of the software products that were used. It also included verification that only approved modules were used in the analyses. In general, the EMCF documentation with the checklist does not provide a sufficient record of what was checked and how it was reviewed. If an independent review, such as the one performed by the NRC, identifies errors or technical issues there is not a mechanism to trace back and determine why the error may have occurred. As will be discussed below, the NRC did identify errors and technical issues. It is natural that there may be errors or omissions in an evaluation that is as large as the WMA C PA. The review process is designed to correct hopefully all, but most likely most, of the errors. DOE does not preserve the checking records for the NRC to verify the thoroughness of the checking process.

The first part of the data validity review is the transparency and traceability of the information that was used. DOE provided references (usually next to the data) that allowed the NRC to trace the data through the analyses. NRC was able to follow the information through multiple levels of references and evaluate modifications to the data that were performed. DOE's documentation of data sources was clear and complete. Review effort (either internal or external) can be lessened if the original source of data is provided as the reference, though this is not required. As long as data sources can be traced through the documentation chain then data transparency and traceability can be verified, and it was for this review.

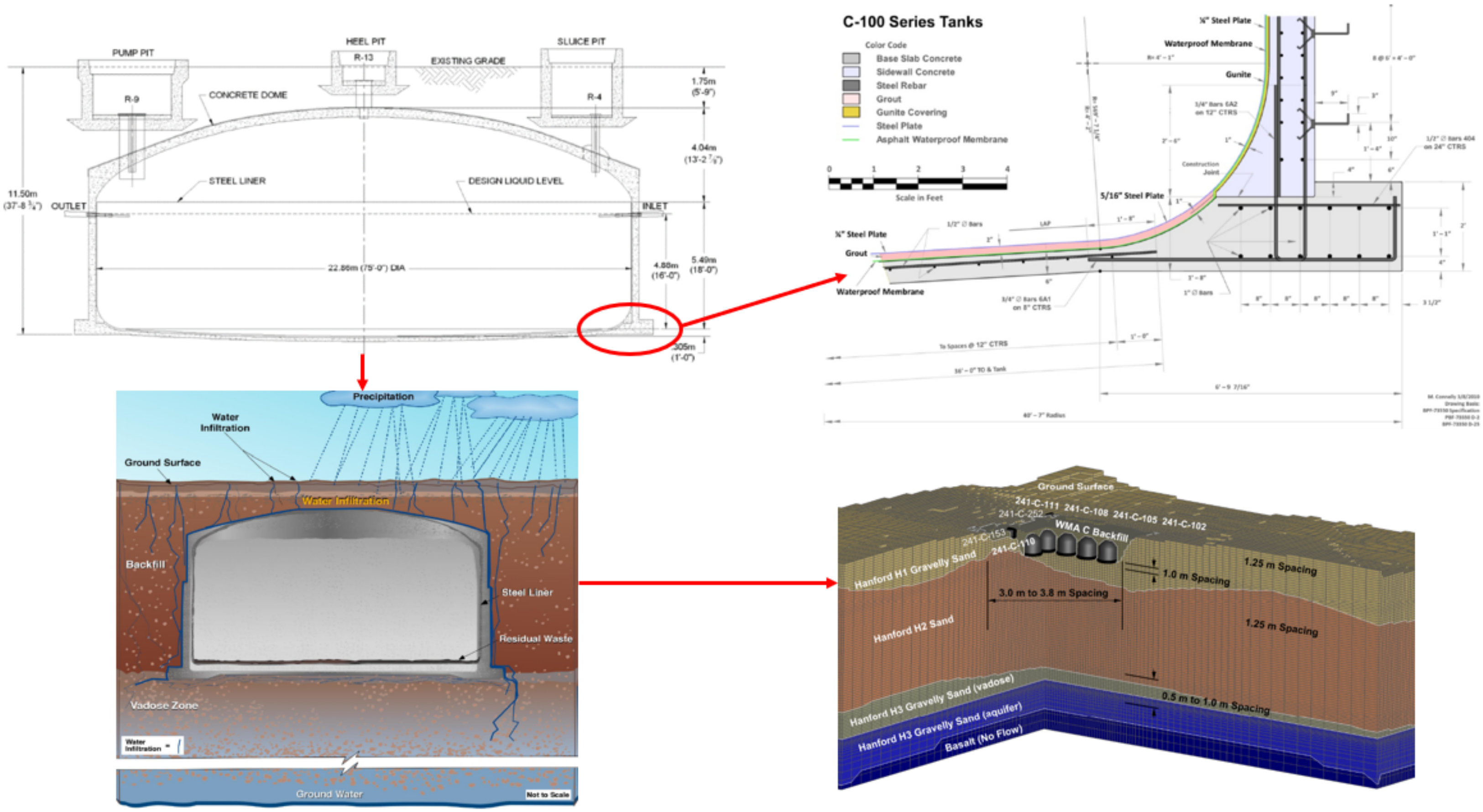

Figure 3-40 Overview of Model Development Information (DOE, 2016)

Some sources of data can be more complex, such as the flow-field outputs from the STOMP modeling. This type of data, input and output files of process models, can be included as appendices to documents but in some cases may be too large or not useful. In these cases, it is useful to provide an appendix that lists the file names and instructions for the reviewer as to how to obtain the files.

The second part of the QA requirements for data validity is that the data has been qualified for use in the analyses. Performance assessments rely on a large amount of information and it is common that information from other programs and literature are used in the analyses, as was the case for the WMA C PA. Much of the data used in the PA modeling was not collected under a quality assurance program at Hanford but came from external sources. Data may be qualified by comparison to other data, or by documentation and evaluation such as by independent peer review. DOE uses quality assurance procedures when collecting or measuring data at their sites. External data that is widely cited and used usually has sufficient quality assurance, but not always. Sometimes when data is traced back to the source, it is found that the data may not have undergone an appropriate quality assurance review. NRC experienced this situation with geochemical data widely that was used throughout the industry. A contractor to NRC traced the geochemical data to its source and when they reviewed the source information, they identified errors in the widely used publication (ML083240260). NRC's review of data used in the PA is found in the respective sections of this report. DOE could better ensure the quality of the data used in the PA by implementing a formal process to review, characterize, and document the qualification of external data used in the PA.

DOE provided information throughout their reports, especially in the PA documentation, to describe the numerical development process. DOE could better demonstrate the sufficiency of the numerical model in terms of temporal and spatial discretization to simulate the real-world systems. DOE's EMCF on the STOMP model was a good attempt to examine the discretization (numerical dispersion) in the STOMP model (RPP-CALC-60448). However, DOE did not have a similar evaluation to demonstrate why the elimination of the discrete features and complexities associated with the systems was appropriate, especially for near-tank modeling. As noted in the model support section, the tanks have inlet ports that act as pathways for advection in the present and they are not going to be sealed. DOE has not selected a grout formulation and demonstrated that a shrinkage gap will not form at the tank periphery. These discrete features could be important and yet the base case numerical model uses coarse discretization with uniform properties and diffusion-only release.

In addition to reviewing the higher-level documents (such as the Draft WIR Evaluation and the performance assessment), the NRC staff reviewed a large number of additional references. NRC placed emphasis on reports that described historical operation and incidents or events. As discussed in NRC's review of model support, confidence in the numerical models could be enhanced if a larger effort to evaluate, synthesize, and compare historical information to the numerical models were undertaken. A summary or listing of historical observations and discrete features of the system followed by a description of how those items were included in the conceptual and numerical models (or why they did not need to be included) would make a stronger case as to the quality of the modeling effort.

NRC issued RAI 2-1 on DOE's QA applied to the WMA C PA (NRC, 2019). NRC had identified discrepancies and inconsistencies in documentation, inputs, or other aspects of the calculations.

None of the items identified, if corrected, change the conclusions of the analysis; most were minor from a risk-significance standpoint. The reason NRC raised these issues was to facilitate the risk-informed review process, which relies on the risk being calculated correctly. It was also to determine if QA assurance processes possibly need to be enhanced for future waste evaluations. As discussed previously, these items are tied to the checking and review process documented in the EMCFs. If some of the steps of the model review process are only captured on a checklist, then the root cause of the problem cannot be determined when errors are identified during independent review. DOE agreed with NRC that some items were errors. DOE felt other items that NRC characterized as errors were modeling decisions (DOE, 2019). Based on the discussions NRC had with DOE staff and their contractors, NRC would characterize the following items as errors because they were not known to DOE until after the NRC review.

- The doses for the acute intruder from the pipelines source were reported as 0.36 mSv (36 mrem). DOE indicated the pipeline was assumed to be 5 percent full of waste. The dose from a pipeline that is 5 percent full of waste should have been 0.018 mSv (1.8 mrem).
- A portion of the WMA C source term was modeled as not being covered by the final closure cap.
- For analyses cases where the tank materials were assumed to be replaced and water was to flow through the source term, water was being routed around the source term instead of through it because of the choice of hydrologic properties.
- Modeling of radionuclide transport in the system model used flow fields calculated with STOMP. The system model was one-dimensional though the system being modeled is three-dimensional. DOE selected a node to abstract velocities from that was under the center of the tank. The velocities are extremely low in this location because of the tank shadow effect. Though this could be a modeling decision, it is a non-conservative assumption that is not supported and should have been questioned during the internal review process.

As previously stated, NRC evaluated these items and determined they would not invalidate the conclusions of the analysis. However, if DOE did not have significant margin to work with between the results and the performance objectives these items could have been significant. For this reason, NRC recommends that the review and checking process be enhanced with more time afforded to the reviewers and a more complete record of the checking process produced (Recommendation #57).

<u>Summary of Review</u>

- The NRC staff reviewed DOE's implementation of quality assurance. DOE has quality assurance procedures and provided records to document the model development process. DOE had high-quality processes for data transparency and traceability. DOE also had robust procedures for software configuration management and control. DOE provided records for the checking and review process, though some records were limited considering the complexity of the evaluation. Overall the quality assurance was sufficient for the Draft WIR Evaluation.

- There were no significant uncertainties associated with quality assurance.

- Recommendation #57 discussed in this section is also included in Table 5-1 of this report.

3.16 NRC Conclusions for Criterion B

The NRC staff evaluated DOE's demonstration that Criterion B of DOE Order 435.1 would be met with reasonable expectation. DOE's demonstration that Criterion B would be met was based on computational models and supporting reports and information. The NRC staff reviewed the computer models, the supporting reports, hundreds of reference reports, and ancillary calculations and files. In addition, the NRC staff had a series of public teleconferences to discuss DOE's analyses prior to NRC developing their RAIs (NRC, 2019).

The NRC staff determined that the standards DOE applied to demonstrate compliance with DOE Order 435.1 were comparable to the requirements found in 10 CFR Part 61. In addition, DOE's specification 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. For waste residuals, site selection is not part of the evaluation, 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 control.

Table 3-19 provides DOE's safety functions and the NRC staff's evaluation of the safety functions. Based on the NRC staff's review of the information submitted, a risk significance is assigned to each safety function and the strength of the technical basis DOE provided is classified as high, moderate, or limited. In some cases, designs or actions were not yet completed by DOE at the time of the NRC staff review, therefore, the technical basis could not be classified. For some safety functions, DOE elected not to take credit and, therefore, provided limited information on the topic. DOE described why they did not take credit for some safety functions.

Table 3-19 Safety Functions Used in the PA and NRC's Assessment of Safety Functions

Safety Function Designation	Description	NRC Assessment of Safety Functions
I1: **Institutional control**	By DOE Order 435.1, Radioactive Waste Management, it is assumed that control of the site will be retained for 100 years. A strong potential exists that the U.S. government will retain control of the site for a much more extended period of time.	**Risk Significance**: Not applicable. It is relevant to the safety of the facility, but it isn't part of the conceptual model. This institutional safety function is an event determined by regulation: At 100 years after closure, reliance on institutional control ends.
I2: **Societal memory**	Societal memory is represented by records, deed restrictions, and other passive controls so that there is a general awareness that the disposal has occurred at the site. DOE Order 458.1 requires record keeping that would lessen the likelihood of the disposal site being forgotten.	**Risk Significance**: Not applicable. Not relevant to the outcome of the PA although it is relevant to the safety of the facility. This institutional safety function is not a feature or a barrier that can be evaluated in the PA but is an event determined by regulation: At 100 years after closure, societal memory of the site as a place of disposal can no longer be relied on.
I3: **Exposure location**	By DOE Order 435.1, it is assumed that a post-closure well is established 100 m downgradient at the point of highest exposure. It is unlikely that this situation will occur, and exposures would be more likely to occur further downgradient.	**Risk Significance**: Not applicable. Not relevant to the outcome of the PA although it is relevant to the safety of the facility. This institutional safety function is not a feature or a barrier that can be evaluated by the PA, but an event determined by regulation: At closure, a hypothetical well and therefore the point of calculation will be located 100 m (328 ft) downgradient of the site's fenceline.

Safety Function Designation	Description	NRC Assessment of Safety Functions
S1: Site characteristics	The existence or absence of various features, events, and processes that give the WMA C site more advantages as a disposal site than other potential locations (e.g., low precipitation rates or thick unsaturated zone).	**Risk Significance**: Not applicable. Safety function S1 is a set of many features related to the characteristics of the chosen site and mostly accounted for in the PA, with the possible except of the precipitation process, under the vadose zone or the saturated zone safety functions. If this safety function was associated exclusively with the precipitation process, which in turn effects the infiltration and recharge rates, the risk significance would have to be high.
EB1: RCRA cover (infiltration reduction)	The final design cover is projected to produce low rates of infiltration for 100's of years.	**Risk Significance**: Low **Technical basis**: Final design of the cover not yet available. Although not significant to total system performance due to other redundant barriers, preliminary cover designs shown in the WMA C PA provide information to determine a RCRA cover could reduce the infiltration rates so as to limit the amount of water available to contact and transport radioactive waste during the first 500 years after closure.
EB2: RCRA cover (depth of disposal)	Limitation of types of potential inadvertent human intrusion by thickness of the final design cover or depth of disposal.	**Risk Significance**: High **Technical basis**: Final design of the cover not yet available. Preliminary cover designs shown in the WMA C PA demonstrate that a RCRA cover could maintain a depth to waste to provide protection to an inadvertent intruder for most structures. Impacts to intruders from excavation would exceed the performance objectives for numerous structures.
EB3: Steel shell (permeability)	The carbon steel shell could limit water flow through the tank.	**Risk Significance**: Not applicable. Not relevant to the outcome of the PA although it is relevant to the safety of the facility. By choice of the DOE, safety function EB3 is not part of the PA and information or a technical basis that supports its capability as a safety function was not provided.
EB4: Steel shell (chemical)	The carbon steel shell will corrode and potentially slow radionuclide transport.	**Risk Significance**: Not applicable. Not relevant to the outcome of the PA although it is relevant to the safety of the facility. By choice of the DOE, safety function EB4 is not part of the PA and information or a technical basis that supports its capability as a safety function was not provided.

Safety Function Designation	Description	NRC Assessment of Safety Functions
EB5: Tank structure (structural)	The dome and walls provide structural support preventing subsidence of the closed facility.	**Risk Significance**: Low **Technical basis**: Moderate. Not significant to total system performance due to other redundant barriers (i.e., the grout). However, the NRC staff review determined that DOE provided reasonable assurance in the WMA C PA that sufficient structural support will exist to prevent significant subsidence within the WMA C so as to meet the performance objective §61.44.
EB6: Tank structure (intrusion)	The tank structure provides a barrier to intrusion.	**Risk Significance**: High **Technical basis**: High to moderate. There is not quantitative technical information as to the effectiveness of the types of engineered components associated with the 100- and 200-series tanks to deter modern drilling technologies from penetrating them for 100's of years in the future. The NRC staff agrees it is reasonable to assume these materials will deter drilling for enough time that the radioactivity will decay.
EB7: Tank structure (chemical)	The concrete shell of the tank acts to condition the chemistry, with sorption characteristic of high pH environments.	**Risk Significance**: Moderate **Technical basis**: Sorption to the basemat reduces the flux of radionuclides to the environment. DOE used literature values and does not know the quality/condition of the basemat concrete.
EB8: Tank structure (permeability)	The concrete of the tank structure is substantially intact and provides a barrier to flow into the tank.	**Risk Significance**: High **Technical basis**: Limited. Low hydraulic conductivity values assigned to the tank structures limit advective release of waste to the environment. However, little information was provided on the long-term impacts of the rebar and exposed steel. DOE should test plausible alternative conceptual models based on steel degradation that disrupts the tank structure and increases the hydraulic conductivity and effective diffusion coefficients assigned to the tank structure.

Safety Function Designation	Description	NRC Assessment of Safety Functions
EB9: Grout in tank (permeability)	The grout acts to limit water flow through the facility, making releases from the waste diffusion-dominated.	**Risk Significance**: Low under low flow conditions, otherwise High. **Technical basis**: Final grout formulation not yet available. NRC strongly encourages DOE to test plausible alternative conceptual models based on the spatial and temporal performance of each safety functions related to features with cementitious material, these include EB8 - Tank structure (permeability); EB9 - Grout in tank (permeability); and EB13 - Tank Base Mat (permeability).
EB10: Grout in tank (chemical)	The grout acts to condition the chemistry, with sorption characteristic of high pH environments.	**Risk Significance**: Low **Technical basis**: Final grout formulation not yet available. In tank grout is currently not credited for impacting release rates with the exception of uranium.
EB11: Grout in tank (structural)	The grout provides structural support preventing subsidence of the closed facility.	**Risk Significance**: High **Technical basis**: Strong. Without a cement-based grout, the tank structures would eventually deteriorate, and subsidence would occur. The NRC staff review determined that there is reasonable assurance that a cement-based grout can maintain sufficient structural support to prevent significant subsidence within the WMA C so as to meet the performance objective §61.44.
EB12: Grout (intrusion)	The structural strength of the grout provides a barrier to intrusion.	**Risk Significance**: Low unless EB6 is not effective **Technical basis**: Grout formulation is to be established. There is not quantitative technical information as to the effectiveness of the types of engineered components associated with the 100- and 200-series tanks to deter modern drilling technologies from penetrating them for 100's of years in the future. The NRC staff agrees it is reasonable to assume these materials will deter drilling for enough time that the radioactivity will decay.

Safety Function Designation	Description	NRC Assessment of Safety Functions
EB13: Tank basemat (permeability)	An intact tank basemat provides a barrier that will limit flow and contaminant transport.	**Risk Significance**: High **Technical basis**: Limited Low hydraulic conductivity values of the tank basemats limit advective release of waste residuals. However, little information is provided on the long-term presence of the rebar. DOE should test plausible alternative conceptual models based on eventual rebar degradation and increased hydraulic conductivity and effective diffusion coefficients of the tank basemat.
EB14: Tank basemat (chemical)	The concrete basemat is anticipated to continue to provide a high pH environment, with associated sorption, for an extended time in the future.	**Risk Significance**: Medium **Technical basis**: Moderate. The sorption of radionuclides to the basemat reduces fluxes to the environment. Most radionuclides are expected to experience sorption to concrete. Uncertainty is associated with the quality of the concrete and its condition.
EB15: Pipelines (permeability)	Intact pipelines provide a delay to releases of waste.	**Risk Significance**: Not applicable. Not relevant to the outcome of the PA although it is relevant to the safety of the facility. By choice of the DOE, safety function EB15 is not part of the PA and information or a technical basis that supports its capability as a safety function was not provided.
AP1: Grout (air pathway)	Limitation of releases to air owing to low air permeability and long pathway to the surface.	**Risk Significance**: Low **Technical basis**: Moderate. Grout reduces the rate of transport of gaseous species to the environment if there are no fast pathways (e.g., shrinkage gap). Since the grout formulation is not established the performance of the grout cannot be fully established.
WF1: Residual waste (chemical)	The residual waste is recalcitrant by nature, providing limitations to the amount and rate of release of contamination.	**Risk Significance**: Low **Technical basis**: High. The impact of the waste to reduce releases is only credited for ^{99}Tc and uranium isotopes. The bases were developed from laboratory experiments on actual waste.

Safety Function Designation	Description	NRC Assessment of Safety Functions
VZ1: **Thickness of the vadose zone**	The vadose zone is thick leading to long travel times.	**Risk Significance**: High under anticipated climate conditions **Technical basis**: Moderate. The NRC staff assumes that safety function VZ1 includes the travel time within the length of the unsaturated zone flowpath and that this time is dependent of the pore-water velocity. In the WMA C PA, DOE showed that within 10,000 years, pore-water velocity in the unsaturated zone was the most important barrier for delaying the arrival time of peak concentrations. Risk significance of this feature would decrease proportionately with higher precipitation rates. If current climatic and hydrogeologic conditions do not significantly change, alternative conceptual models of the vadose zone performed such that the results did not appreciably diverge from the results of the base case model used in the WMA C PA. NRC staff determined that base case values associated with the unsaturated Hanford H2 sand unit have an adequate technical basis and support.
VZ2: **Sorption on vadose zone soils**	Vadose zone soils sorb some of the contaminants of potential concern, delaying their arrival at the water table.	**Risk Significance**: High **Technical basis**: Moderate. The use of the constant soil adsorption coefficient, or adsorption-desorption distribution coefficient (Kd) model, was assumed by DOE to be generally applicable in the WMA C PA when contaminants are present at low concentrations. The values assigned were based on laboratory studies, testing, and measurements. Appropriate Kd values were assumed with the exception of a few radionuclides. The NRC staff has determined that the performance of this unsaturated zone safety function has an adequate technical basis and support.
VZ3: **Dispersion in the vadose zone**	Spreading of contaminants in the vadose zone, dispersing them and decreasing concentrations.	**Risk Significance**: Medium **Technical basis**: Moderate. Variation in velocity results in dispersion of solute mass within the vadose zone that affects the magnitude of the peak groundwater dose. However, predicted velocities are relatively low. If current climatic and hydrogeologic conditions do not significantly change, alternative conceptual models of the vadose zone produced results that did not appreciably diverge from the results of the base case model used in the WMA C PA. NRC staff determined that base case values associated with the unsaturated Hanford H2 sand unit have an adequate technical basis and support.

Safety Function Designation	Description	NRC Assessment of Safety Functions
SZ1: Water flow in saturated zone	Advective flow in the saturated zone leading to dilution of the contaminants.	**Risk Significance**: High **Technical basis**: Limited. This safety function is one of the most important. The key component of the average flux is the layer thickness weighted hydraulic conductivity value. The NRC staff has concluded that a stronger technical basis is needed for using the relatively high hydraulic conductivity value assigned by DOE for the unconfined aquifer. There are several sources of uncertainty associated with this value which is the basis for the amount of water available for dilution of contaminant concentrations.
SZ2: Sorption on saturated zone soils	Saturated zone soils sorb some of the contaminants of potential concern, delaying their arrival at the point of compliance.	**Risk Significance**: Low **Technical basis**: Weak to Moderate. The use of the constant soil adsorption coefficient, or adsorption-desorption distribution coefficient (Kd) model, was assumed by DOE to be generally applicable in the WMA C PA when contaminants are present at low concentrations. Values assigned were based on laboratory studies, testing, and measurements. Appropriate Kd values were assumed with the exception of a few radionuclides. Although DOE a stronger technical basis for unsaturated zone Kd values, the NRC staff has determined that the performance of this safety function has an adequate technical basis and support. No uncertainty, barrier importance, or sensitivity analysis demonstrated a strong risk significance associated saturated zone Kd values.
SZ3: Dispersion in the saturated zone	Spreading of contaminants in the saturated zone, dispersing them and decreasing concentrations.	**Risk Significance**: Medium to High **Technical basis**: Weak to Moderate. Dispersion in saturated zone is ranked relatively high in importance in the WMA C PA. NRC staff has concluded that a stronger technical basis is needed for the assigned longitudinal field-scale dispersivity value for the saturated zone.
SZ4: Dilution in well	Dilution caused by pumping a groundwater well.	**Risk Significance**: Not applicable. Not relevant to the outcome of the PA although it is relevant to the safety of the facility. This safety function is not a feature or a barrier to evaluated by DOE in the PA and determined by regulation.

Explicit assumptions associated with the NRC staff conclusions are provided in the text. In the NRC staff's professional judgment, those assumptions are likely to be essentially confirmed. In the event of significant deviations, the conclusions found below may no longer be valid. Unless explicitly stated, the remaining uncertainties associated with demonstrating that other criteria will be met are not found to be significant with respect to the conclusions provided here.

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

- NRC's conclusions with respect to compliance with 10 CFR Part 61.42 did not take credit for the probability of intrusion.
- The closure date will be a minimum of 2050.
- The climate and hydrogeological system will remain relatively constant with current conditions.
- The existence and impact of the Columbia River dams upstream of the Hanford Site will remain unchanged.
- The assumed volume and composition of waste in components yet to be characterized will be verified through implementation of characterization plans.
- DOE will implement a design for grout that has minimal shrinkage, verified by testing of proper-scale or other means.
- The thick layers of grout, reinforced concrete, and steel will deter intrusion, such as drilling, for up to 500 years in the future.
- Engineered surface cover will be designed and subsequently perform as described.

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

- DOE has demonstrated compliance with 10 CFR 61.41 for all components.
- DOE has demonstrated compliance with 10 CFR 61.42 for the tanks and ancillary equipment, except the plugged pipelines.
- DOE has not demonstrated compliance with 10 CFR 61.42 for plugged pipelines because of the large uncertainties associated with the radiological composition of waste remaining in the pipelines.
- DOE has demonstrated compliance with 10 CFR 61.43.
- DOE has demonstrated compliance with 10 CFR 61.44.

The recommendations provided in Section 3 (Recommendation #8 through #57) are collated into Table 5-1. These recommendations are categorized as (1) applicable to the Draft WIR Evaluation for WMA C, (2) consider for future evaluations for waste management areas, or (3) general technical recommendations that would generally improve the basis for the technical information but are not essential to the evaluation for WMA C.

4 CRITERION C – Assessment of Radionuclide Concentrations and Classification

This section documents the materials DOE submitted and the NRC staff's review of the information with respect to the assessment of radionuclide concentrations and the resultant waste classification of the residual waste in WMA C. Waste classification is a tool to help determine the suitability of radioactive waste for near-surface disposal and ensure waste is managed properly. The review entailed evaluation of the physical form of the waste and radionuclide concentrations and classification of the waste.

4.1 Waste Physical Form

The third criterion in DOE Manual 435.1-1 (referred to here as Criterion C) provides that wastes:

> …[W]ill 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.

In the Draft WIR Evaluation for WMA C, DOE indicated that stabilized tanks, ancillary structures, and residuals (waste) would be incorporated into a solid physical form and would be covered by a closure barrier. DOE provided waste classification calculations using information about the volume and radioactivity of the waste residuals developed for the PA calculations to demonstrate that the waste residuals would be Class C or below.

In the Draft WIR Evaluation for WMA C, DOE provided an initial approach to waste classification. The NRC staff asked questions about the DOE approach in clarification calls and then documented those questions in RAIs (NRC, 2019). DOE revised their approach in response to the RAI (DOE, 2019). The NRC staff reviewed the DOE approach as supplemented by their responses. Review of the revised approach is emphasized in the following sections, and the initial approach is summarized.

4.1.1 DOE's Assessment that Waste will be Incorporated into a Solid Physical Form

Residual waste in WMA C is in different forms. Much of the waste is solid or sludge but some of the waste is liquid. As described above, the DOE Manual 435.1-1 criteria specify that the wastes must be incorporated into a solid physical form. The basis for this requirement is that solid wastes are much less likely to be dispersed into the environment from disturbance processes or from natural degradation processes affecting the engineered barriers used to retain waste. NRC's LLW disposal regulations state that liquid waste must be solidified or packaged in sufficient absorbent material to absorb twice the volume of the liquid. In addition, solid waste containing liquid shall contain as little free standing and noncorrosive liquid as is reasonably achievable, but in no case shall the liquid exceed 1 percent of the volume.

In the Draft WIR Evaluation for WMA C, DOE stated that waste will be incorporated into a solid physical form but did not indicate how that was going to be achieved. The NRC staff asked DOE to describe the quantity of residual liquids remaining in the systems at closure and how those liquids will be incorporated into a solid physical form (NRC, 2019). In response, DOE provided the following information (DOE, 2019).

DOE acknowledged that at the end of waste retrieval operations, some liquids that were not pumpable would remain within WMA C. Liquids can be present mixed with the solids to form sludge or be present as free liquids (e.g., supernate). Table 4-1 is the volume of liquids for each component. Most of the volume estimates are from DOE's inventory report (RPP-RPT-42323, 2015). However, some volumes are estimates (e.g., for the C-301 tank and CR-Vault) based on the assumption that 90 percent of the current volume will be removed when retrieval begins for those components. The cells within the CR-Vault are periodically pumped with the last reported volumes from 2010.

In its response to the NRC staff's RAI, DOE indicated that the design process for grout formulation and placement includes accounting for the total amount of liquid in the system, whether added to the grout or existing within the residual waste. When grout will be placed in tanks or other structures, all liquid remaining within the residual wastes, along with the free water associated with grout preparation and placement, will be either be absorbed within the grout as the cement hydrates or will be evaporated by the heat of hydration. The resulting grouted mass will be a solid physical form. DOE indicated that a supporting calculation is being prepared to document the basis for this conclusion, but that it was not yet completed at the time the RAI response was developed. DOE later provided the calculation to the NRC (DOE, 2020, ADAMS Accession No. ML20042C425).

DOE stated that other ancillary equipment in WMA C, such as pits, diversion boxes, pipelines and encasements are sloped and designed to drain to tanks. No appreciable liquids are expected to remain even in pipelines that are plugged. Waste residuals remaining in these components will be solids. Pits, diversion boxes and some encasements will be grouted to prevent subsidence; pipelines will not be grouted except for the extent to which grout may flow into them incidentally as connected structures are filled.

4.1.2 NRC Evaluation that Waste Will be Incorporated into a Solid Physical Form

The NRC staff evaluated DOE's information provided in the response to the RAI supporting that waste will be incorporated into a solid physical form. As shown in Table 4-1, the total volume of liquids projected to remain in the system is 71.9 m^3 (19,000 gal). This represents about 19 percent of the total volume of residual waste evaluated in the PA.

Though NRC's waste characteristics requirements do not apply to DOE, the NRC staff compared DOE's approach to achieving a solid physical form against the requirement for commercial LLW disposal. The commercial LLW requirement is that solid waste containing liquid shall contain as little free standing and noncorrosive liquid as is reasonably achievable, but in no case shall the liquid exceed 1 percent of the [total] volume. There are two components to the requirement, a "reasonably achievable" aspect and a 1-volume percent aspect.

As indicated previously, the total volume of liquids projected to remain in the system is 71.9 m^3 (19,000 gal). Though this volume represents about 19 percent of the total residual waste volume, it represents a much smaller fraction of the system void volume when grouted/stabilized. It is this latter fraction that is useful to compare with the commercial LLW requirement as it is intended to be applied to solidified waste. The volume of the solidified waste cannot be large to arbitrarily decrease the liquid percentage.

Table 4-1 Estimated Liquid Remaining in WMA C Components at Closure

Tank / Cell	Residual Liquids (Gallons) [1]
C-101	845
C-102	2588
C-103	247
C-104	1272
C-105	177
C-106	85
C-107	5297
C-108	232
C-109	890
C-110	1287
C-111	890
C-112	3366
C-201	2
C-202	2
C-203	15
C-204	11
C-301	1140
CR-001	520
Cell 1	2
CR-002	27
Cell 2	1
CR-003	170
Cell 3	2
CR-011	0
Cell 11	1

[1] To convert to L multiply by 3.78

The total liquid volume represents only about 0.3 percent of the grouted components at closure; it would be well within the industry standard for commercial LLW with no further liquid removal. On an individual component basis, only three components would be projected to be over 1% by volume free liquids: C-301 (3%), CR-001 (1.3%), and CR-003 (1.7%). The values for these components are based on assumptions and will be revised as waste retrieval and sampling is completed for these components. Free liquids are mainly a concern for potential impacts to groundwater, and these systems have relatively low volumes of residual waste.

DOE is developing their grout formulation and the grout placement process to account for free liquids in the tanks. DOE plans to take actions in the closure process to reduce, to the extent practical, remaining free liquids in tanks. For ancillary equipment, DOE indicated that the ancillary equipment is designed to drain into tanks or sumps such that free liquids will be minimal. The NRC staff agrees with this conclusion for most of the ancillary equipment. However, the NRC staff does not agree with this conclusion for plugged pipelines. It is not known where the pipelines are plugged (or even if they remain plugged). Only if a pipeline was plugged at the upgradient end of the line would it be anticipated that the line was drained of liquids. If the line plugged in any downgradient point, liquids would be expected to be present behind the plug. Grouting of pipelines is not part of DOE's closure plans. Without characterization data, the NRC staff believe it is not appropriate to conclude that the plugged lines have had the residual waste incorporated into a solid physical form.

The grout quality and the residual waste volume that may remain as free liquids will be strongly dependent on the grout formulation, its placement process, and curing conditions within the tanks. Ventilation may need to be provided and adequate curing time between pours will be essential. DOE has grouted tanks at other sites within the DOE complex; therefore, DOE has experience in this area. The unique composition of the waste and the geometrical configuration of the tanks and access points may produce differences from the experiences at other sites. DOE will need to verify and validate that the grout formulation will be able to achieve the design goals. As DOE has done for other projects, large-scale cold testing may provide the technical basis for the grout formulation and placement process to ensure the grout is of high-quality. Once grouting is performed, the tanks will have reached an end state that is relatively irreversible. The NRC staff agrees that, after proper testing, if DOE implements their design for the grout, including the placement and curing process, the tanks will have residual wastes incorporated into a solid physical form to the extent practical.

Summary of Review

- The NRC staff reviewed DOE's assessment that waste will be incorporated into a solid physical form. The NRC staff agrees that, after proper testing, if DOE implements their design for the grout, including the placement and curing process, the tanks will have residual wastes incorporated into a solid physical form to the extent practical. Without characterization data, it is not accurate to conclude that the plugged lines have had the residual waste incorporated into a solid physical form.

- The significant sources of uncertainty are performance of the final grout design to adsorb liquids and the amount of liquids in plugged pipelines.

- There are no recommendations associated with this section.

4.2 Radionuclide Concentrations and Classification

Criterion C states that incidental waste managed as LLW must 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 information that DOE submitted, and the NRC staff's review of that information, follows.

4.2.1 DOE's Assessment of Radionuclide Concentrations and Classification

Most of the residual waste at WMA C will be left in engineered structures (e.g., reinforced concrete, steel); only the waste residuals contained in piping will not be contained in robust engineered structures, though the pipes themselves will provide a barrier to release for some period of time. Many of the pipes are inside encasements that provide secondary containment. DOE plans to install a closure cover to limit long-term infiltration of water to the residual waste, to protect the waste residuals from release to the environment by natural processes, such as erosion or biointrusion, and to decrease the likelihood that an inadvertent intruder will contact the residual waste in the future after institutional control is no longer maintained. The closure cover will be designed to ensure there is at least 5 m (16 ft) of material between the land surface and buried waste residuals.

NRC's waste classification tables, Tables 1 and 2 of 10 CFR 61.55, were developed assuming an intruder unknowingly excavated into the buried waste (underneath a 2 m (7 ft) thick cover) to install a basement for a residence after the end of the institutional control period (100 years). Due to the disposal depth of the stabilized tanks, ancillary structures and residuals at WMA C, DOE indicated the basement excavation exposure scenario would not be appropriate for WMA C. Instead, a more appropriate and credible exposure scenario would be to evaluate a hypothetical intruder who drills a well through a waste tank or ancillary structure after the assumed institutional control period. DOE assumed that the institutional control period would be 100 years (in the Draft WIR Evaluation for WMA C), although DOE indicated that it anticipates the institutional control will continue well beyond that period. Hypothetical intrusion was assumed to occur at 100 years after closure for the pipelines and 500 years after closure for all other structures, since the other structures will be filled with a cement-based grout and, therefore, will be more highly stabilized.

Initially, DOE used an approach to waste classification based on NRC guidance provided in NUREG-1854 (NRC, 2007). NUREG-1854 outlines several approaches to waste classification for incidental waste, starting with simple and progressing to more complex. The simple approaches involve mass- or volume-based averaging of waste concentrations within a structure, with or without taking credit for materials added to stabilize the waste. The complex approaches involve adjustments based on the site-specific differences of the waste, disposal configuration, and the disposal site, relative to what was assumed when the waste classification tables in 10 CFR Part 61.55 was developed.

DOE indicated that they used a site-specific averaging approach (Category 3 from NUREG-1854) that is risk-informed by accounting for the conditions of the site, the final form of the stabilized residuals, site-specific parameters and the final closure configuration. DOE developed averaging expressions based on the inadvertent intruder analysis performed for WMA C. DOE used the total inventory of residual waste within each tank or ancillary structure and assumed the waste was spread in an even layer over the plan view area of the structure. The residual waste within pipelines was assumed to be spread evenly over the internal surface of the pipelines. The averaging expressions calculated the ratio of the concentrations of residual waste (radionuclides) to the values provided in Table 1 or 2 of 10 CFR 61.55. The resultant numbers were multiplied by a "site factor" that was derived from the site-specific intruder analyses dose results and the assumption that the Table 1 or 2 concentrations in 10 CFR 61.55 were equivalent to an inadvertent intruder dose of 5 mSv (500 mrem). The analyses accounted for the different levels of difficulty in disturbing the waste residuals – the piping could be disturbed at 100 years whereas the other structures were assumed to not be disturbed until 500 years after closure.

The waste classification procedure provided in 10 CFR Part 61 utilizes a sum-of-fractions (SOF) approach. The SOF is used to account for the contribution of individual radionuclides. If the SOF is less than 1.0, then the waste is Class C or less. Long-lived and short-lived radionuclides are considered separately. The largest SOF DOE calculated was 0.0297 for Tank C-107.

The NRC staff reviewed the initial DOE approach and commented in the RAI (see Section 4.2.2). In response, DOE revised their waste classification calculations (DOE, 2019). DOE developed revised averaging expressions for acute (Equation 1) and chronic (Equation 3) inadvertent intruders who were assumed to intrude at either 100 or 500 years after closure, depending on the structure.

$$SOF_i = \frac{C_{Ri}}{Table_{Valuei}} * \left(\frac{Waste_{thickness}}{Drill_{depth}}\right) * \left(\frac{Exposure_{drill}}{Exposure_{NRC}}\right) * \left(\frac{1}{0.254 * 0.5}\right) \qquad Eqn. 1$$

Where:

SOF_i = Radionuclide “i” contribution to the sum of fractions

CR_i = Concentration of radionuclide “i” at closure (i.e., assumed 2068 closure date) decayed 400 years for all tanks, C-301, CR-Vault tanks and cells, and no decay after closure for the pipelines, valve pits, and diversion boxes (Ci/m^3 or nCi/g)

$Table_{Valuei}$ = Class A concentration limit from 10 CFR 61.55 Table 1 or Table 2 (radionuclide “i”)

$Waste_{thickness}$ = thickness of the residual waste for radionuclide “i” (m)

$Drill_{depth}$ = total depth of the well at WMA C Tank Farm (m)

$Exposure_{drill}$ = time of exposure for the WMA C PA acute drilling scenario (hours)

$Exposure_{NRC}$ = time of exposure for the NRC acute excavation scenario (hours)

0.254 = NRC dilution factor assumption for Class C intruder analysis – areal mixing of excavation material and waste (dimensionless)

0.5 = NRC dilution factor assumption for Class C intruder analysis – waste barrels on 50% full of waste (dimensionless)

and

$$C_{Ri} = \frac{I_{Ri}}{V_w} \; or \; \frac{I_{Ri}}{M_w} \qquad Eqn. 2$$

I_{Ri} = Inventory of radionuclide “i” at closure (i.e., assumed closure date of 2068) decayed 400 years for all tanks, C-301, CR-Vault tanks and cells, and no decay after closure for the pipelines, valve pits, and diversion boxes (Ci or nCi)

V_w = Residual waste volume (m^3)

M_w = Residual waste mass (g)

Equations 1 and 2 adjust the calculated concentrations relative to the concentrations in the regulation for the dilution factor and exposure time appropriate for deeper buried waste (i.e., a drilling exposure scenario compared to an excavation exposure scenario). Because 100 years of decay was assumed for Class A waste (e.g., shallowly buried waste or waste without a robust intruder barrier), the concentrations were not decayed for the pipelines, valve pits, and diversion boxes but were decayed an additional 400 years for the tank residuals and CR-244 vault. Equation 3 was developed for the chronic post-drilling exposure scenario:

$$SOF_i = \frac{C_{Ri}}{Table_{Valuei}} * \left(\frac{V_{w,drill} / V_{T,drill}}{V_{w,NRC} / V_{T,NRC}} \right) * \left(\frac{1}{0.254 * 0.5} \right) \qquad Eqn. 3$$

SOF_i = Radionuclide "i" contribution to the sum of fractions

CR_i = Concentration of radionuclide "i" at closure (i.e., using the assumed 2068 date discussed previously) decayed 400 years for all tanks, C-301, CR-Vault tanks and cells, and no decay after closure for the pipelines, valve pits, and diversion boxes (Ci/m^3 or nCi/g)

$Table_{Valuei}$ = Class A concentration limit from 10 CFR 61.55 Table 1 or Table 2 (radionuclide "i")

$V_{w,drill}$ = volume of waste brought to the surface from drilling (m^3)

$V_{T,drill}$ = total volume of soil brought to the surface from drilling (m^3)

$V_{w,NRC}$ = volume of waste brought to the surface from NRC excavation scenario (m^3)

$V_{T,NRC}$ = total volume of soil brought to the surface from NRC excavation scenario (m^3)

0.254 = NRC dilution factor assumption for Class C intruder analysis – areal mixing of excavation material and waste (dimensionless)

0.5 = NRC dilution factor assumption for Class C intruder analysis – waste barrels on 50% full of waste (dimensionless)

The parameters DOE used in the averaging expressions were provided in Tables 16 to 22 of DOE's response to the NRC staff's RAI (DOE, 2019). Initially, DOE did not classify all structures. In their revised calculations they included the tanks within the CR-Vault, Tank C-301 and ancillary equipment. Table 4-2 provides the results of DOE's revised waste classification calculations. All components were estimated to be less than Class C. The plugged pipelines and the tanks within the CR-Vault were the most limiting. The plugged pipelines had a SOF of 1.0 whereas the tanks in the CR-Vault had SOFs that ranged from 0.46 to 0.80. Overall, the acute results were more limiting than the chronic results. The Table 1 radionuclides (long-lived) were limiting for the CR-Vault tanks whereas Table 2 radionuclides (short-lived) were limiting for the plugged pipelines.

NRC allows for alternative methods for waste classification using the requirements in 10 CFR 61.58. The approach in 10 CFR 61.58 is based on an evaluation of the specific characteristics of the waste, disposal site, and method of disposal that demonstrates with reasonable assurance that the performance objectives in Subpart C of 10 CFR Part 61 will be met. DOE provided a comparison of the results of the averaging expressions to the intruder results of the PA assuming a SOF of 1.0 was equivalent to 5 mSv (500 mrem). The ratio of the averaging expression results to the PA intruder doses ranged from 8 to 11. Because the ratios are greater than 1, this indicates the averaging expressions are conservative.

4.2.2 NRC Evaluation of the Assessment of Radionuclide Concentrations and Classification

Before describing the NRC staff's review of DOE's waste classification calculations for WMA C, it is important to provide some background material on NRC's waste classification system. The background material will allow the reader to better understand the review comments. The starting point for waste classification is to understand the development of NRC's waste concentrations provided in 10 CFR 61.55. The waste classification system was developed to allow commercial radioactive waste generators to classify their waste on a waste package basis to provide protection of inadvertent intruders. The primary assumption was that the waste would be buried within 3 m (10 ft) of the land surface such that an inadvertent intruder could unknowingly develop a residence on the disposal facility at 100 years or longer after the disposal facility is closed. More dangerous waste (i.e., Class C) would be buried more deeply or with a robust intruder barrier such that the waste could not be excavated into for at least 500 years. NRC evaluated a variety of potential intruder scenarios, but the regulations were based primarily on two exposure scenarios: acute construction and chronic residence. The acute construction receptor was a person involved in the excavation and building of the residence which was assumed to take 500 hours. The chronic resident receptor was a person who lived in the residence after it was built. Doses were calculated for each of these receptors and the concentrations that would result in either a 5 mSv (500 mrem) dose to the whole body or a limiting dose to other organs (roughly equivalent radiological risk to a 5 mSv (500 mrem) whole body dose) were determined. Then the limiting value for each receptor type for each isotope considered was determined. In the rulemaking process, numerous modifications were made to these initially calculated concentrations.

The documents describing the development of the waste classification tables in 10 CFR 61.55 are large, numerous, and sometimes conflicting. It is potentially a difficult task for DOE or other stakeholders to understand some of the information. The NRC staff have begun working on tools to increase regulatory efficiency and decrease regulatory burden associated with waste classification, but those tools are not yet available (anticipated 2021) (Ridge et al, 2019).

Primarily in response to public comment on the draft rule, the concentrations initially calculated by the NRC were increased by a factor of 10 to account for the assumption that not all waste disposed in a facility would be at the waste concentration limit. For ^{137}Cs, the calculated concentration was increased an additional factor of 20. Other modifications also occurred. For instance, the concentrations calculated for long-lived TRU radionuclides were different for different isotopes but were simplified in the final regulation to a common value of 0.37 Bq/kg (10 nCi/g) for Class A waste. Some of these assumptions, however, while valid for disposal of commercial LLW may not be applicable to waste classification of WIR wastes.

The Table 1 and 2 waste concentrations were developed assuming an intruder excavation exposure scenario with calculation of potential doses to both acute and chronic receptors. Acute doses were dominated by inhalation and direct radiation exposure pathways whereas chronic doses were dominated by ingestion of contaminated plants and animals. The more limiting concentration to each receptor type (acute or chronic) was used to develop the regulation. Radioactive decay for 100 years was incorporated into the Class A values and 500 years was incorporated into the Class C values. Table 4-3 summarizes the modifications and key assumptions employed for development of the 10 CFR Part 61.55 waste classification tables (Table 1 and 2 of 10 CFR 61.55). As can be seen from examining Table 4-3, the concentrations found in Table 1 and 2 of 10 CFR 61.55 are not directly comparable to a 5 mSv (500 mrem) whole body dose without first correcting for the modifications that were made to the initially calculated values. The guidance provided in Appendix B of NUREG-1854 (NRC, 2007) described the process to develop the averaging expressions. The averaging expressions were to be used as a review tool by the NRC staff.

Table 4-2 Summary of DOE Class C SOF Results

Equipment	Acute Class C Equation		Chronic Class C Equation	
	Table 1 SOF	Table 2 SOF	Table 1 SOF	Table 2 SOF
C-101	$2.8x10^{-2}$	$1.0x10^{-4}$	$6.3x10^{-3}$	$2.3x10^{-5}$
C-102	$7.4x10^{-2}$	$1.5x10^{-4}$	$1.7X10^{-2}$	$3.3x10^{-5}$
C-103	$7.9x10^{-3}$	$6.8x10^{-5}$	$1.8X10^{-3}$	$1.5X10^{-5}$
C-104	$1.0x10^{-2}$	$7.1x10^{-5}$	$2.4X10^{-3}$	$1.6X10^{-5}$
C-105	$7.3x10^{-2}$	$2.8x10^{-4}$	$1.6X10^{-2}$	$6.4X10^{-5}$
C-106	$4.9x10^{-2}$	$4.4x10^{-4}$	$1.1X10^{-2}$	$9.9X10^{-5}$
C-107	$2.3x10^{-2}$	$1.5x10^{-4}$	$5.1X10^{-3}$	$3.5X10^{-5}$
C-108	$1.1x10^{-3}$	$1.2x10^{-5}$	$2.6X10^{-4}$	$2.8X10^{-6}$
C-109	$6.0x10^{-4}$	$2.2x10^{-5}$	$1.3X10^{-4}$	$5.0X10^{-6}$
C-110	$1.2x10^{-3}$	$2.5x10^{-5}$	$2.8X10^{-4}$	$5.5X10^{-6}$
C-111	$6.2x10^{-3}$	$3.8x10^{-4}$	$1.4X10^{-3}$	$8.6X10^{-5}$
C-112	$1.6x10^{-2}$	$6.1x10^{-4}$	$3.6X10^{-3}$	$1.4X10^{-4}$
C-201	$2.7x10^{-1}$	$2.5x10^{-5}$	$6.1X10^{-2}$	$5.7X10^{-6}$
C-202	$2.4x10^{-1}$	$4.4x10^{-5}$	$5.3X10^{-2}$	$1.0X10^{-5}$
C-203	$8.0x10^{-3}$	$2.1x10^{-5}$	$1.8X10^{-3}$	$4.7X10^{-6}$
C-204	$4.7x10^{-4}$	$1.4x10^{-5}$	$1.1X10^{-4}$	$3.1X10^{-6}$
C-301	$3.8x10^{-1}$	$4.4x10^{-4}$	$8.5X10^{-2}$	$9.8X10^{-5}$
CR-001	$4.6x10^{-1}$	$5.3x10^{-4}$	$1.0X10^{-1}$	$1.2X10^{-4}$
CR-002	$4.1x10^{-1}$	$4.7x10^{-4}$	$9.2X10^{-2}$	$1.1X10^{-4}$
CR-003	$6.9x10^{-1}$	$8.0x10^{-4}$	$1.6X10^{-1}$	$1.8X10^{-4}$
CR-011	$8.0x10^{-1}$	$9.2x10^{-4}$	$1.8X10^{-1}$	$2.1X10^{-4}$
Pits	$1.5x10^{-3}$	$2.3x10^{-2}$	$3.4X10^{-4}$	$5.3X10^{-3}$
Boxes	$1.5x10^{-3}$	$2.3x10^{-2}$	$3.3X10^{-4}$	$5.1X10^{-3}$
Pipelines	$3.4x10^{-3}$	$5.2x10^{-2}$	$7.7X10^{-4}$	$1.2X10^{-2}$
Plugged Pipelines	$6.8x10^{-2}$	$1.0x10^{+0}$	$1.5X10^{-2}$	$2.4x10^{-1}$

The guidance noted the assumptions associated with development of 10 CFR 61.55, however, did not explicitly provide what the assumptions were to allow DOE or other external stakeholders to adjust their values accordingly. The constants developed for the averaging expressions in NUREG-1854 are simplified for use as a review tool (e.g., a different constant would be derived for each radionuclide, but the most limiting value was determined and applied to all radionuclides).

There are three primary considerations when evaluating the classification of residual waste. First, what is the concentration and quantity of residual waste remaining in each component? Second, how and when is residual waste disturbed? And third, what is the concentration of waste when it reaches the environment and people can interact with it? The NRC staff reviewed DOE's initial waste classification calculations and found that DOE did not properly account for the assumptions made in developing the waste concentration tables in 10 CFR 61.55 for application to commercial LLW disposal. DOE did not initially classify all the components that may be left in place and used the average concentration of waste for some components (NRC, 2019). In response to NRC's comments, DOE revised their approach (DOE, 2019).

Table 4-3 Key Assumptions and Modifications for Development of 10 CFR 61.55

Assumption or Modification	Description
Waste is buried shallow	Excavation exposure scenario is appropriate. For deeper buried waste, modifications to the exposure scenarios are necessary.
Institutional control period of up to 100 years. For Class C waste 500 years of decay.	For time periods other than 100 or 500 years, the concentrations of radionuclides will need to be decayed.
Concentrations of all radionuclides increased by a factor of 10 to account for waste not being at the concentration limit in commercial LLW disposal.	For WIR determinations, need to divide Table 1 and 2 concentrations by 10, except for ^{137}Cs.
Concentration of Cs-137 increased by a factor of 20 to account for waste not being uniformly at the concentration limit in commercial LLW disposal.	For WIR determinations, need to divide the Table 2 concentration of ^{137}Cs by 20.
Impacts to acute (construction) and chronic (residence) calculated, most limiting value used for waste classification.	Because the most limiting value for two exposure scenarios was used, results of an acute intruder or chronic intruder analysis will not be directly comparable.
Radionuclides assessed were based on anticipated waste streams at the time.	Some radionuclides important to WIR are not included in Table 1 and Table 2 of 10 CFR 61.55.
Waste streams not anticipated to contain large amounts of both short-lived and long-lived radionuclides.	Regulation states to not combine results for Table 1 and Table 2 radionuclides to determine classification. However, if WIR waste streams do contain large amounts of both short-lived and long-lived radionuclides, they should be combined.

To understand the similarity of residual waste to LLW from a hazard standpoint, the NRC staff first calculated the concentrations of radionuclides in the different system components and calculated the SOF. The most limiting component the DOE identified for long-lived radionuclides was the C-011 tank within the CR-244 Vault (because it had the thickest waste layer). The SOF is 352 for the C-011 tank, based solely on what is in the waste layer. Therefore, based solely on the radioactivity in the waste layer, the waste would be Greater-Than-Class-C (GTCC) waste. However, the waste is not in thick layers in a shallow trench (i.e., what NRC's waste classification system was based upon) but rather is in relatively thin layers, deeper underground with overlying grout and steel. It is appropriate to account for these differences when classifying the waste. The most important aspect to consider is protecting public health and safety (offsite and onsite) and that will be driven by the concentrations present when or if exposures were to occur, which will be considerably less than the concentrations in the waste under the ground.

As described in Section 4.2.1, DOE developed revised averaging expressions in consultation with the NRC. The revised averaging expressions are designed to "back out" the key assumptions and replace them with the relevant site-specific information. For example, for waste buried deeper than 5 m (17 ft) below the land surface, a drilling exposure scenario is appropriate to consider. The excavation depth assumed in the NRC analysis was 3 m (10 ft).

For the acute intruder, DOE developed Equation 1. The NRC staff's review of Equation 1 is summarized below:

- This equation starts with the Class A concentrations in Table 1 and 2 of 10 CFR 61.55 ($Table_{valuei}$). It is appropriate for DOE to use the Class A concentrations because 100 years of institutional control were assumed for Class A waste, the same as for the pipelines and some of the other ancillary equipment that are not grouted (e.g., do not have a robust intruder barrier).
- DOE calculated C_{Ri}, the concentration of each radionuclide I, by using the concentration at the assumed closure time (year 2068) for those components that would not be grouted and calculating an additional 400 years of decay and ingrowth for those radionuclides present in components that would be grouted. This is consistent with the NRC approach as a total of 500 years of decay and ingrowth was included in the development of the radionuclide concentrations for Class C waste in 10 CFR 61.55. DOE's calculation of the concentrations of radionuclides is appropriate.
- DOE developed a drilling dilution factor as the ratio of the average thickness of each waste layer divided by the depth to the resource (i.e., 79 m (260 ft) – water).
- DOE accounted for the difference in the exposure time of the acute driller compared to the acute home constructor (40 hours compared to 500 hours).
- DOE included two additional factors of 0.254 and 0.5. DOE described the 0.254 factor as the NRC dilution factor assumption for Class C intruder analysis – areal mixing of excavation material and waste (dimensionless). This factor is a volumetric dilution factor, not an areal mixing factor, and 0.254 can be calculated from the volumes provided in the reports but 0.25 was used in the IMPACTS codes by NRC. In addition, the 0.25 was not applied to the acute intruder. It was assumed the concentrations to which the acute intruder was exposed were not mixed. This factor should not be included in Equation 1, as it is making DOE's acute SOF estimates too high (i.e., conservative) by roughly a factor of 4. For plugged pipelines, the uncertainty in the inventory is greater than this factor of 4 conservatism for acute intruder waste classification.
- DOE included a factor of 0.5 for what they describe as an NRC dilution factor assumption for Class C intruder analysis – waste barrels are 50 percent full of waste. This factor was to account for a packing efficiency of waste in the facility, not that individual barrels were 50 percent full of waste. Nonetheless, it is appropriate to include this factor in the denominator.
- For some radionuclides, the calculated concentrations are per unit volume while for the long-lived TRU radionuclides, the concentrations are per unit mass. DOE used the same density for all components (2.05 g/m^3) which is at the upper end of the range observed (1.2 to 2.1 g/m^3) from sampling and characterization of tank residuals. If the average density observed was used, the SOF DOE calculated would have been above 1.0 for the limiting components. In future evaluations, DOE should use measured densities (Recommendation #58)

The highest SOFs DOE calculated for the acute intruder was 1.0 for the plugged pipelines for short-lived radionuclides and 0.8 for the CR-011 tank for long-lived radionuclides.

DOE developed Equation 3 to determine the classification of residual waste for the chronic intruder. The primary difference between Equation 3 and Equation 1 is that Equation 3 does not include differences in exposure time and it also uses a different approach to calculate the mixing or dilution of residual waste compared to the acute intruder. The NRC staff's review of Equation 3 is summarized below. Only differences between the equations are noted (e.g., the NRC staff's comments on the development of the terms $Table_{valuei}$ and C_{Ri} for the acute intruder apply to the chronic intruder):

- DOE provided a term ($V_{W,NRC}/V_{T,NRC}$) to back out the NRC dilution factor for the chronic intruder. This term is the 0.254 (0.25) value provided at the end of the equation. It was included twice, making DOE's SOF too high by a factor of 4 with respect to this factor.
- DOE included a term ($V_{W,drill}/V_{T,drill}$) to account for a drilling dilution factor for the chronic intruder. This denominator of this factor was calculated as the area of the suburban gardener's lot (2,500 m^2 [27,000 ft^2]) multiplied by a plowing depth of 0.15 m (0.49 ft). Use of this factor is effectively assuming that the whole property is plowed/tilled prior to use. The application of this mixing depth for the whole property is likely to be non-conservative. The contribution of radionuclides that impact the inhalation and external dose exposure pathways are likely to be underestimated.

In future evaluations, DOE should use adjusted equations as described in this review (Recommendation #59). DOE calculated the most limiting SOF to the chronic intruder of 0.18 for the CR-011 tank from long-lived radionuclides and 0.24 for the plugged pipelines from the short-lived radionuclides. In determining if the WMA C components are GTCC waste, the NRC staff believes adjustments are appropriate for the chronic intruder calculations.

The assumption of the 15 cm (0.49 ft) mixing depth for the cuttings is central to the chronic intruder waste classification calculations. It is appropriate to use a mixing depth of 15 cm (0.49 ft) for the garden portion of the property. The contributions to SOF that DOE calculated from radionuclides that contribute primarily through the plant ingestion pathway (e.g., ^{90}Sr) appropriately used a 15 cm (0.49 ft) cuttings depth. However, for the bulk of the property, the question becomes how much of the cuttings are mixed with natural materials after the cuttings are placed on the land surface. There will be some mixing as the property is developed and aeolian erosion and deposition occur. However, it is not expected that most residential properties are plowed prior to establishing residence. Some grading is expected around the home foundation, but wells usually have a required offset distance from the foundation of a home. A mixing depth of 1 cm (0.39 in) is more appropriate for the bulk of the property where crops are not raised. This is consistent with foot traffic, movement of equipment, planting a yard, and similar activities.

The NRC staff examined the dose results in DOE's PA model for the chronic intruder. Depending on the component and time period examined, the non-plant ingestion pathways accounted for a maximum of approximately 20-30% of the dose. The smaller dilution factor would only apply to this fraction of the dose contribution. When the extra factor of 4 is backed out and the extent of the plowing depth of 15 cm (0.49 ft) is adjusted to 1 cm (0.03 ft) for the non-plant ingestion pathways, the NRC staff calculated that the SOF for the chronic intruder would be 0.17 for the CR-011 tank compared to DOE's value of 0.18.

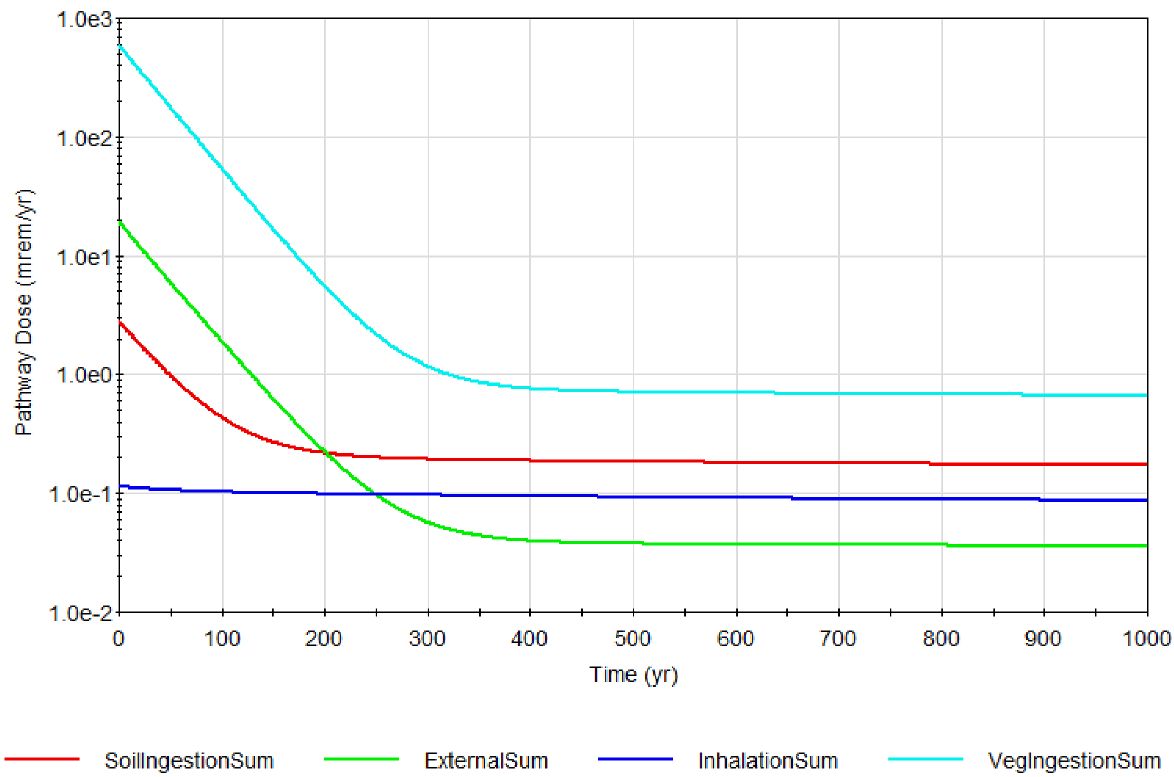

Figure 4-1 Chronic Intruder Doses for the CR-Vault

Essentially, the extra factor of 4 from double counting of the NRC dilution factor is offset by the assumption regarding cuttings mixing. Figure 4-1 is the intruder dose plot for the chronic intruder for the 244-CR vault. The doses drop off very rapidly with time from the decay of ^{90}Sr and ^{137}Cs. Doses during the first 100 years are not possible due to active institutional controls being in place.

As discussed in Section 3.5.1, DOE accounted for uncertainty in the residual waste inventory by propagating uncertainty in the volume, radionuclide concentrations, and density. When DOE propagated the uncertainties, the overall uncertainty in the inventory of each radionuclide ranged from approximately a factor of 2 or less above the mean and a factor of 2 or less below the mean. Waste classification is performed assuming the concentrations are known. However, the inventory of residual waste is an uncertain estimate based on limited sampling and numerous assumptions. The uncertainty in the radionuclide concentrations should be considered when classifying the waste. NRC does not expect waste generators to know the exact inventory of a canister of waste before classifying it according to 10 CFR Part 20, Appendix G. Uncertainty is inherent in a large inventory of canisters that are disposed in a facility. The NRC staff reviewed DOE's uncertainty estimates and found that the approach to calculating RSDs may have underestimated the overall uncertainty in the inventory. The NRC staff estimated the uncertainty in inventory as approximately double what DOE estimated. However, even with this uncertainty, the NRC staff believes that all the components except the plugged pipelines would be Class C waste or less.

As discussed in Section 3.5.2, the inventory remaining in the plugged pipelines is considerably more uncertain than the other components in the system. As a result of no waste being retrieved from a plugged pipeline, they are classified as having the highest SOF based on assuming the concentration of radionuclides in the pipelines are represented by the average concentration of waste remaining throughout WMA C. Inventory remaining in a plugged pipeline could be representative of what was being transferred at the time of plugging (i.e., a discrete event) or it could represent a gradual buildup of material in the pipeline over time followed by eventual plugging. The records of waste transfers are incomplete, potentially with 25 percent of the transfers not accounted for.

DOE has not performed sampling of a plugged pipeline. The complex processes that lead to plugging of a pipeline results in high uncertainty for the inventory remaining in the pipe. The inventory uncertainty would be expected to be larger than that shown in Figure 3-18 of Section 3.5.2, which is up to ± 2 orders of magnitude on an individual radionuclide basis. Though the expected waste classification for a plugged pipeline is Class C, the actual waste classification could be considerably less or considerably higher. DOE should characterize plugged pipelines to determine the concentration of radionuclides and the amount of free liquids that are present (Recommendation #60).

For deeper buried waste, it is appropriate to consider a drilling exposure scenario. The total amount of waste and soil mixture exhumed during a drilling event at the Hanford Site may only be on the order of 1 m^3 (3 ft^3). For the excavation exposure scenario assumed for development of NRC's waste classification system, 680 m^3 (24,000 ft^3) of soil was assumed to be exhumed along with 232 m^3 (8,200 ft^3) of waste and backfill. A portion of the exhumed mixture was assumed to be used to backfill around the home foundation. The remainder was assumed to be spread on a normal-sized lot. If the volume of waste and soil exhumed is only on the order of 1 m^3 (3 ft^3), there is likely not enough material to support the assumption that it is spread over the whole building lot or that it persists indefinitely. If the layer of cuttings is very thin, it would be susceptible to mixing with other deposited natural materials and erosion. The NRC staff has begun evaluating dose assessment exposure scenarios for drilling (Esh et al, 2020). Preliminary conclusions are that the current approaches to dose assessment for the chronic intruder are likely to be conservative. This means that waste classification performed based on a chronic intruder driller exposure scenario are also likely to be conservative.

When the waste classification tables (i.e., Table 1 and 2 of 10 CFR 61.55) were developed, the limiting concentration from acute and chronic exposure scenarios for each radionuclide was used. This can add additional conservatism to the classification based on acute and chronic receptors separately, as DOE has done. For example, DOE used the Class A limit for ^{90}Sr, which is 1,480 MBq/m^3 (0.04 Ci/m^3), for both the acute and the chronic receptor. The Class A limit is based on the chronic exposure scenario for ^{90}Sr. In the calculations performed in 1981 to create the Class A limit, the calculated value for ^{90}Sr in the acute exposure scenario was 66,200 MBq/m^3 (1.79 Ci/m^3). For most radionuclides the differences are not this large, but for many important radionuclides to classification of residual wastes, the differences in the limiting concentrations for the acute and chronic exposure scenario can be a factor of 2 to 3. For mixtures of radionuclides, some of the radionuclides may be classified based on concentrations that are more restrictive than were calculated in the intruder assessment to develop the regulation. Presently, it is difficult for stakeholders to account for this impact in a waste evaluation. The NRC staff is working on developing tools to allow stakeholders to more efficiently investigate the methods used to create the waste tables.

As mentioned in the summary of DOE's waste classification, NRC allows for alternative methods for waste classification using the requirements in 10 CFR 61.58. The approach in 10 CFR 61.58 is based on an evaluation of the specific characteristics of the waste, disposal site, and method of disposal that demonstrates with reasonable assurance that the performance objectives in subpart C of 10 CFR Part 61 will be met. DOE provided a comparison of the results of the averaging expressions to the intruder results of the PA assuming a SOF of 1.0 was equivalent to 5 mSv (500 mrem). The ratio of the averaging expression results to the PA intruder doses ranged from 8 to 11. As described above, in both Equation 1 and Equation 3 an extra factor of 4 was included in the DOE calculations. If that factor of 4 is removed, the averaging expressions would result in a factor of 1.5 to 2.8 higher values than the DOE intruder dose assessment. This is generally good agreement considering all the potential sources of differences in these types of calculations (e.g., dose conversion factors using other International Commission on Radiological Protection methodologies). Because the revised ratios are greater than 1, this indicates the averaging expressions are conservative with respect to DOE's intruder dose assessment but not by an excessive margin.

Summary of Review

- The NRC staff reviewed DOE's assessment of radionuclide concentrations and classification of the residual waste. The NRC staff agrees that the residual waste in WMA has been properly classified as Class C or less, except for the plugged pipelines. The uncertainty in the inventory remaining in the plugged pipelines is too large to allow for accurate waste classification.

- The significant source of uncertainty is the inventory remaining in plugged pipelines.

- Recommendations #58, #59, and #60 are found in this section and are summarized in Table 5-1.

4.3 NRC Conclusions for Criterion C

The NRC staff evaluated DOE's demonstration that wastes will be incorporated into a solid physical form. Staff evaluated the basis and calculations to demonstrate that the concentrations of radionuclides in the waste does not exceed the applicable concentration limits for Class C LLW as set out in 10 CFR 61.55.

The NRC staff performed a risk-informed review of the information provided. Assumptions associated with the NRC staff conclusions are provided in this section. In the NRC's professional judgment, those assumptions are likely to be confirmed by DOE as further technical work is performed and if eventual closure of WMA C occurs. In the event of significant deviations in actual conditions from assumed conditions, the conclusions found below may no longer be valid. Unless explicitly stated, remaining uncertainties associated with demonstrating that other criterion will be met are not found to be significant with respect to the conclusions provided here.

The following assumptions apply to the conclusions for waste classification:

- Sampling and characterization of the inventory within the C-301 catch tank and the tanks in the CR-244 vault will be performed and will verify the assumptions made about the inventory of those components.
- The WMA C closure date is 2068.
- Active institutional controls will end 100 years after closure.
- The NRC staff conclusions are not based on an approach relying on 10 CFR 61.58.

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

- DOE has demonstrated that residual waste remaining in all components within WMA C and within the scope of this review, with the exception of plugged pipelines, has been or will be incorporated into a solid physical form.
- DOE has not demonstrated that residual waste remaining in plugged pipelines has been or will be incorporated into a solid physical form.
- DOE has demonstrated that the residual waste remaining in all components within WMA C and within the scope of this review, with the exception of plugged pipelines, will have concentrations of radionuclides that does not exceed the applicable concentration limits for Class C radioactive waste.
- Because of the uncertainty in the inventory of plugged pipelines and lack of characterization data, DOE has not demonstrated that the residual waste remaining in plugged pipelines within WMA C will have concentrations of radionuclides that do not exceed the applicable concentration limits for Class C radioactive waste.

The recommendations provided in Section 4 (Recommendation #58 through #60) are collated into Table 5-1. These recommendations are categorized as (1) applicable to the Draft WIR Evaluation for WMA C or (2) consider for future evaluations for waste management areas.

5 OVERALL NRC REVIEW RESULTS AND CONCLUSIONS

The NRC staff has completed a risk-informed, performance-based review of the Draft WIR Evaluation for WMA C, the WMA C PA, and supporting documents and information. The Hanford Site and the residual waste are very complex and, because the site has been operated for a long period of time, 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 all the information the NRC staff considered using a standard of reasonable expectation and comparison to the criteria in DOE Manual 435.1-1.

The NRC staff concludes the following, <u>in all WMA C tanks and ancillary equipment except the plugged pipelines</u>:

- DOE has 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 has 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 has 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 NRC staff concludes the following <u>for the plugged pipelines</u>:

- As a result of not having characterization data, the uncertainty in the inventory of plugged pipelines is too large. DOE has not demonstrated that it meets the above criteria for the plugged pipelines. The NRC recommends that DOE characterize the plugged pipelines to determine the concentration of radionuclides and the amount of free liquids that are present.

In addition, the NRC staff notes that DOE has indicated it plans to complete the following activities, which are necessary to validate the assumptions DOE makes in the Draft WIR Evaluation for WMA C and the WMA C PA:

1) DOE indicated that they will complete waste retrieval and characterization of waste from the C-301 catch tank and the CR-244 Vault to verify the assumptions made in the Draft WIR Evaluation. The NRC staff agrees this characterization is necessary to more accurately understand the inventory of material remaining.
2) DOE intends to complete the final closure cover design and verify its performance, including an evaluation of erosion protection. The NRC staff agrees this closure cover

design is needed because the closure cover can hinder inadvertent human intrusion into the residual waste (e.g., by excavation) for hundreds of years after closure.

3) DOE plans to select the final grout formulation and verify that it can achieve the necessary performance (i.e., confirm that it will have no shrinkage, will not degrade significantly over the period of analyses, and verify that the grout will have the target effective diffusion coefficients and hydraulic conductivities for the field-scale materials at high water-to-cement ratios). The NRC staff agrees with this approach because the final grout formulation will play a key role in limiting water contact with the waste and in limiting the release of radioactivity to slow diffusional release.

The NRC staff has provided recommendations throughout this report for Criteria A, B, and C. Many of those recommendations were not risk-significant for WMA C but could be important to consider when developing waste evaluations for other waste management areas. Table 5-1 provides a listing of the recommendations in this TER and identifies them as (1) applicable to the Draft WIR Evaluation for WMA C, (2) consider for future evaluations for waste management areas, or (3) general technical recommendations that would generally improve the basis for the technical information but are not essential to this evaluation for WMA C. Recommendations that are categorized as “consider for future evaluations” could be risk-significant depending on the specific details of the waste management area being evaluated, whereas, “general technical recommendations” are simply noted as best practices for performing waste evaluations.

NRC does not have regulatory authority over the waste management decisions made by the DOE for incidental waste at Hanford. DOE does not have an obligation to implement recommendations provided by the NRC.

Table 5-1 Recommendations for the DOE Based on NRC's Review of the Draft WIR Evaluation for WMA C

Number	Recommendation	Section	Applicability#
1	The isotopes ^{242}Cm and ^{94}Nb should be added as key radionuclides	2.1 – Identification of Key Radionuclides	Future evaluations
2	Use of a tiered approach to volume-based retrieval combined with risk insights may allow for reduced impacts to the public with fewer resources expended.	2.2 – Removal to the Maximum Extent Practical	Future evaluations
3	The criteria for terminating retrieval of waste by chemical means should be adjusted based on field experience when field experience differs from laboratory experience.	2.2 – Removal to the Maximum Extent Practical	General
4	When sluicing and pumping are the primary removal technology, the number of access locations (sluicing points) should be a primary consideration and additional access points should be created if necessary.	2.2 – Removal to the Maximum Extent Practical	General
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	General
6	Mechanical robotics solutions for the Hanford site's remaining tank farms should be more thoroughly considered.	2.2 – Removal to the Maximum Extent Practical	General
7	The requirement to reach the limit of technology for the first technology used before moving to a second (or third technology) should be reconsidered. This approach may result in an inefficient use of resources and ultimately higher risk (given budget and schedule constraints).	2.2 – Removal to the Maximum Extent Practical	Future evaluations
8	DOE should follow guidance for DOE Order 435.1 when evaluating potential peak dose impacts.	3.1 – Assessment Context	General
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	Future evaluations
10	The safety case approach to model development using results of uncertainty and sensitivity analyses should be clearly described and documented.	3.2 – Future Scenarios and Conceptual Models	Future evaluations

Number	Recommendation	Section	Applicability#
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	Future evaluations
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	Future evaluations
13	The range for recharge rates applied to long-term, non-disturbed conditions should be expanded.	3.3 – Current Climate and Recharge	Future evaluations
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	Future evaluations
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	Future evaluations
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	WMA C
17	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	WMA C
18	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	Future evaluations
19	Degradation of exposed steel and rebar that impact the properties of walls and basemats should be evaluated.	3.4 – Engineered Barrier System	Future evaluations

Number	Recommendation	Section	Applicability#
20	The basemats should be assumed to be free of cracks only if characterization data is available.	3.4 – Engineered Barrier System	Future evaluations
21	Higher basemat effective diffusion coefficients should be used to reflect the uncertainty about construction quality. Sampling and testing of basemat concrete or appropriately analogous materials should be considered.	3.4 – Engineered Barrier System	Future evaluations
22	The performance of safety functions should be tested individually as well as in combination with other safety functions (e.g., grout separation from steel).	3.4 – Engineered Barrier System	Future evaluations
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	Future evaluations
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	Future evaluations
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	Future evaluations
26	The assumptions about the quantity and composition of waste remaining in unplugged pipelines are uncertain. Characterization of some of these pipelines should be completed to determine the quantity and composition of waste remaining.	3.5 – Radionuclide Inventory, Source-term Release, and Near-Field Transport	Future evaluations
27	The inventory in plugged pipelines and encasements should be characterized or conservative estimates of the inventory remaining in pipeline encasements should be developed.	3.5 – Radionuclide Inventory, Source-term Release, and Near-Field Transport	WMA C
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	General

Number	Recommendation	Section	Applicability#
29	DOE should ensure that the effective diffusion coefficient values assigned based on experimental data do not include sorption if it is going to be applied to the basemat layer in the model.	3.5 – Radionuclide Inventory, Source-term Release, and Near-Field Transport	Future evaluations
30	Uncertainty ranges for waste release for the PA should not be limited to the range observed from limited samples. If future empirical testing is performed for waste release, organic compounds present in tank residuals should be included as part of the experimental design.	3.5 – Radionuclide Inventory, Source-term Release, and Near-Field Transport	Future evaluations
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	Future evaluations
32	The modeling of gaseous releases should examine shorter transport pathways if the tank infill grout cannot be designed for zero shrinkage.	3.5 – Radionuclide Inventory, Source-term Release, and Near-Field Transport	Future evaluations
33	Potential lateral movement and transport above the H2 sand unit should be investigated. If model results show this process to be occurring, the significance should be quantified.	3.6 – Flow and Transport in the Unsaturated Zone	Future evaluations
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	Future evaluations
35	The uncertainty ranges for Kd values for ^{79}Se, ^{129}I, ^{226}Ra, and the plutonium isotopes should be expanded.	3.6 – Flow and Transport in the Unsaturated Zone	Future evaluations
36	Documentation of the CPGW model 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	Future evaluations

Number	Recommendation	Section	Applicability#
37	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	Future evaluations
38	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	Future evaluations
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	Future evaluations
40	Saturated zone thickness should be part of a sensitivity analysis.	3.7 – Flow and Transport in the Saturated Zone	Future evaluations
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	Future evaluations
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	Future evaluations
43	A quantitative basis should be developed that engineered components will deter modern drilling methods for 100's of years in the future.	3.10 – Inadvertent Human Intrusion	Future evaluations
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	Future evaluations
45	Consistent approaches to fruit and vegetable ingestion should be used for the onsite and offsite receptors.	3.10 – Inadvertent Human Intrusion	Future evaluations
46	Site-specific values for biosphere parameters should be used when available.	3.10 – Inadvertent Human Intrusion	Future evaluations

Number	Recommendation	Section	Applicability#
47	Measurements of mass loading values that can be assigned to an acute intruder (well driller) should be completed.	3.10 – Inadvertent Human Intrusion	Future evaluations
48	Radon should be included with the dose impacts to the inadvertent intruder.	3.10 – Inadvertent Human Intrusion	Future evaluations
49	When the grout formula has been established and closure conditions are known (e.g., closure barrier design is selected) a structural stability assessment should be completed.	3.12 – Stability of the Disposal Site	WMA C
50	Calculations should be developed to show that potential long-term subsidence is not likely due to leaching of the grout.	3.12 – Stability of the Disposal Site	General
51	The largest volumes of void space associated with pipelines and their encasements should be identified and described.	3.12 – Stability of the Disposal Site	WMA C
52	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	Future evaluations
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	Future evaluations
54	In future uncertainty assessments it should be ensured that the tails of the distributions are not truncated.	3.14 - Uncertainty	Future evaluations
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	Future evaluations

Number	Recommendation	Section	Applicability[#]
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	Future evaluations
57	The review and checking process should be enhanced with more time afforded to the reviewers and a more complete record of the checking process produced.	3.15 – Quality Assurance	Future evaluations
58	Measured densities should be used, when available, for waste classification calculations.	4.0 – Assessment of Radionuclide Concentrations and Classification	Future evaluations
59	Adjusted equations Equation 1 and Equation 3 should be used for future waste classification calculations as described in the NRC review comments.	4.0 – Assessment of Radionuclide Concentrations and Classification	Future evaluations
60	Plugged pipelines should be characterized to determine if liquid remains and what concentrations of radionuclides remain in the plugged pipelines.	4.0 – Assessment of Radionuclide Concentrations and Classification	WMA C

[#] *WMA C* means applicable to the Draft WIR Evaluation for WMA C and is shaded for easy reference; *Future evaluations* means should be considered for future waste evaluations, if necessary, considering the risk significance of the topic to the evaluation; *General* means if completed can improve the technical basis for WMA C or future waste evaluation, and is considered a best practice for performing waste evaluations

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APPENDIX A

Adequacy of the Information Obtained to Address NRC's Request for Additional Information from January 30, 2009

In 1999 and 2003, DOE conducted waste retrieval campaigns for Tank C-106 at WMA C that resulted in an estimated 10.5 m^3 (370 cubic feet) of residual waste remaining in the tank. The HFFACO retrieval criteria for this tank provided in HFFACO Milestone M-045-00 is 10.2 m^3 (360 cubic feet) of residual waste, or the limits of technology, whichever is less. The remaining waste volume in Tank C-106 exceeds the HFFACO volume criteria by an estimated 0.3 m^3 (10 cubic feet). Therefore, the DOE developed an exception request for Tank C-106 and submitted this request to the NRC in October 2004 (ADAMS Accession No. ML051380544) pursuant to Appendix H of the HFFACO, which requires consultation with the NRC when single shell tank retrievals performed under the HFFACO do not meet the milestone M-045-00 criteria after completion of retrieval.

On January 19, 2005, the NRC issued RAIs on Tank C-106 (ADAMS Accession No. ML050070154), to which the DOE responded on August 25, 2005 (ADAMS Accession No. ML052550374). Further, on April 18, 2008, the DOE submitted a revised exception request and the associated documentation to the NRC (RPP-20658, 2008; DOE, 2006). In response to the revised exception request, the staff issued an RAI on January 30, 2009 (ADAMS Accession No.ML090090030), to which the DOE did not respond. A common theme within the RAI suggests that the retrieval exception for Tank C-106 would be more appropriately evaluated within the context of the staff's risk evaluation of the entire WMA C tank farm.

Shortly after NRC issued the RAIs on Tank C-106, DOE changed its approach from evaluating each tank to that of evaluating the entire waste management area (i.e., WMA C). Therefore, DOE did not provide specific responses to the NRC staff's RAIs on Tank C-106, and after discussions between Washington State Department of Ecology, DOE, and NRC, it was agreed that these Tank C-106 responses would be included in DOE's comment response for RAI 2-2 of the RAI responses on the Draft WIR Evaluation for WMA C (DOE, 2019). In letter dated April 23, 2018, NRC closed the previous reviews of Tank C-106 such that closure of the tank could be evaluated in a consistent risk context with other facilities of WMA C (ADAMS Accession No. ML18074A207).

In the "Path Forward" section in the Tank C-106 RAI, Comment 1, the NRC staff provided DOE eight examples, a) through i), of FEPs and asked DOE to discuss their plausibility and their potential significance as alternative conceptual models or alternative future scenarios. Examples a), b), c), and d) from RAI Comment 1 in NRC (2009) provided examples related to climate and the rate of precipitation rate as well as the resulting density and type of vegetation and the rate of erosion. Example a) focused on fire and drought and the potential of the engineered surface cover being without vegetation for certain time periods. Example b) was focused on possible cyclic climate conditions alternating between arid and humid conditions. DOE's modeling used simulations with a steady-state; however, natural processes may not allow steady-state conditions for such long timeframes. Example c) addressed the impacts of a humid, warmer climate may on the assumed long-term, steady-state vadose flow and exclusion of fast pathways, while example d) addressed the potential for a less humid, drier climate to result in potential upward movement of moisture and contaminants in the unsaturated zone being deposited at the surface.

DOE responded to these examples by pointing out that the WMA C PA included a variety of uncertainty and sensitivity analysis conditions with elevated recharge rates including an upper end recharge sensitivity case involving recharge through a gravel surface for the duration of the analysis. Sensitivity case INF03 had the same set of parameters values as the base case, including a recharge value of 0.5 mm/yr (0.02 in/yr) through the intact surface cover; however, the operational period had a 40 percent higher recharge rate (140 mm/yr [5.5 in/yr]) and the post-institutional control period's recharge rate was set equal to 100 mm/yr (4 in/yr). DOE stated that these rates were higher than recharge rates for the base case during a period of no vegetation such as example a). The response to example c) was similar; the range of uncertainty for recharge in the WMA C PA included the potential for climate change, so that the effect of a credible increase in precipitation over the assessment time scale is included in the range of uncertainty evaluated in the WMA C PA. With regards to example c), DOE stated that the range of credible future climate states does not include a climate sufficiently dry to allow upward movement of contaminates, and if such a condition did exist, it would be expected only to enhance the performance of the facility with regard to both air and groundwater. Based on this information, the NRC staff concludes that DOE's responses adequately addressed the comments discussed above.

For example b), DOE again referred to the WMA C PA. DOE stated that the WMA C PA included a variety of uncertainty and sensitivity analysis conditions that address this comment, and that the sensitivity cases included a case outside of the credible range for long-term recharge on natural soils. As for varying climatic conditions changing the influence that FEPs might have on one another (e.g., increased rainfall may increase erosion thereby reducing an engineered cover's thickness and so thereby increasing infiltration), DOE stated that the safety function approach in the WMA C PA is not focused on changes on one parameter or process that may affect another parameter or process but rather on the overall net positive or negative effect on the performance of the safety function. NRC believes certain interdependency and interrelationships between FEPs over time may not be identified by the DOE analysis. In Section 3.2 of this report NRC recommends that DOE refine or improve their hybrid approach to scenario and conceptual model development since DOE's safety function methodology does not appear identify significant interdependencies and interrelationships between FEPs. NRC recommended that DOE emphasize future scenario development and conceptual model development and test for plausible alternative future scenarios. Due to these recommendations, the previous concerns made by NRC staff underpinning the comment made in example b) are captured by the recommendations made in Section 3.2 of this report and thereby superseded.

Example e) discussed the possible changes to concentrations based on changes to the height of the water table. The example stated that future contaminants could end up flowing in different hydrogeological units with differing hydraulic properties based on the height of the water table, thereby causing changes to the dispersion and the linear average velocity of the groundwater in which the contaminants are traveling. DOE stated in their response that while a higher or lower water table could have some effect on the average groundwater flow rate used to approximate the effect of the aquifer dilution safety function, it would still be within the wide range of groundwater flow rates considered in the uncertainty and sensitivity analyses. DOE provided no technical basis supporting this statement and did not include sensitivity cases that tested the effect of the water table height or the thickness of the saturated zone in either the sensitivity analysis or the barrier importance analysis. In Section 3.7 in this report NRC recommended that DOE include saturated zone thicknesses as part of a sensitivity analysis in order to understand the significance of this parameter's capability to influence concentration or dose results. Due to these recommendations, the NRC staff concerns underpinning the comment made in example e) are captured by the recommendations made in Section 3.7 of this report and thereby superseded.

Example f) discussed the impacts of episodic infiltration due to strong rainfall events and associated field evidence. DOE pointed out that episodic infiltration rapidly redistributes in time and space and that an assumed constant infiltration rate applied a few meters below ground surface is appropriate (Oostrom et al., 2016). In addition, this redistribution process is supported by observations of moisture content distributions in the unsaturated zone, and by a controlled field experiment that supported the unsaturated zone modeling approaches used in the PA (Zhang and Khaleel, 2010). Based in this information, the NRC staff concludes that DOE's response has adequately addressed this comment.

Example g) described the possibility of lateral movement of contaminants in the vadose zone in combination with clastic dikes as potential conducts for fast vertical transportation. For example, perched water bodies or semi-saturated lateral movement of contaminants could have the potential to flow along clastic dikes and move relatively quickly to deeper units, thereby bypassing the retarding effects of the vadose zone. The WMA C PA, DOE responded, included extensive analyses evaluating the potential for lateral movement of contamination beneath WMA C in assumed post-closure conditions. RPP-RPT-59197 (2016) documented that the heterogeneous representation of unsaturated zone sediments or the potential occurrence of clastic dikes at WMA C do not result in significant lateral redistribution of contaminant plumes (Section 3.5 of this report discusses this in further detail). In addition, the DOE response in DOE (2019) stated that the lateral movement associated with Tank BX-241 is not relevant to WMA C, as the geological setting is different than at WMA C. Based on this information, the NRC staff concludes that DOE's response has adequately addressed this comment.

In example h) NRC requested additional information with regards to the exclusion of an igneous activity scenario whereby waste is transported by volcanic eruption to a populated zone or whereby the destruction of the grouted tank due to igneous activity would allow contaminants to be transported from the damaged grouted structures by alternative means not being considered currently. In the WMA C PA DOE examined the probability of this event occurring and concluded the potential for this FEP to influence WMA C to have a very small probability of occurrence during the next 10,000 years. In addition, in Appendix H of the WMA C PA DOE discussed FEP 1.2.04 and its potential for changing the infiltration rate as a result of ashfall from regionally significant volcanic activity. The result of this FEP evaluation was that the effect of prior eruptions is included in the paleo record of infiltration which was subsequently captured in the uncertainty range in infiltration. Based in this information, the NRC staff concludes that DOE's response has adequately addressed this comment.

Example i) asked DOE to discuss the justification for exclusion of Native American exposure scenarios from DOE (2006). DOE responded that DOE made a request for Tribal Nation stakeholders to provide exposure scenarios that reflect their traditional activities and that two exposure scenarios were provided. DOE agreed to include quantitative analyses of the Native

American exposure scenarios in the Remedial Investigation/Feasibility Study documents prepared on the Hanford Site. Impacts from Native American exposure scenarios were quantitatively evaluated in a risk assessment of unsaturated zone sediments and soils that were impacted by past tank leaks and releases of waste at WMA C. The exposure scenarios used in this specific risk assessment are documented in RPP-ENV-58813 (2016) and detailed results of those Native American exposure scenario impact evaluations are in RPP-RPT-58329 (2016). NRC recommended in Section 3.8 of this report that DOE include Native American exposure scenarios in future waste evaluations to increase transparency with stakeholders. Based in this information, the NRC staff concludes that DOE's response has adequately addressed this comment.