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Section 6 of 6

Document Information

Document #	DOE/ORP-2005-01	Revision	0
Title	INITIAL SINGLE-SHELL TANK SYSTEM PERFORMANCE ASSESSMENT FOR THE HANFORD SITE		
DATE	04/18/2005		
Originator	CONNELLY MP	Originator Co.	CH2M
Recipient			
References			
Keywords			
Projects			
Other Information			

APPENDIX A

CROSSWALK OF *LOW-LEVEL WASTE DISPOSAL FACILITY*

FEDERAL REVIEW GROUP MANUAL **TO THE**

INITIAL SINGLE-SHELL TANK SYSTEM PERFORMANCE

ASSESSMENT FOR THE HANFORD SITE

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9 **TABLE**

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16 **TERMS**

17 ALARA   as low as reasonably achievable
18 LFRG Manual   *Low-Level Waste Disposal Facility Federal Review Group Manual*
19 LLW   low-level waste
20 PA   performance assessment
21 SST PA   *Initial Single Shell Tank System Performance Assessment for the Hanford Site*
22 SST   single-shell tank

1 **A1.0 CROSSWALK OF *LOW-LEVEL WASTE DISPOSAL FACILITY FEDERAL***
2 ***REVIEW GROUP MANUAL* CRITERIA TO THE *INITIAL SINGLE-SHELL TANK***
3 ***SYSTEM PERFORMANCE ASSESSMENT FOR THE HANFORD SITE***

4 This appendix is taken from *Format and Content Guide for U.S. Department of Energy*
5 *Low-Level Waste Disposal Facility Performance Assessments and Composite Analyses*
6 (DOE 1999a).

Table A-1. Crosswalk of *Low-Level Waste Disposal Facility Federal Review Group Manual* Criteria to the *Initial Single-Shell Tank System Performance Assessment for the Hanford Site* (14 pages)

LFRG Manual Criteria [a]	Content Guide Section and Title [b]	Location in SST PA
Finding I, Criterion 4 – The PA identifies Federal, state, and local statutes, regulations, and agreements that may impact site engineering, facility design, or facility operations. The PA also describes the impacts of those statutes, regulations, and agreements that may be precipitated by the PA results.	Section B.2.4 Related Documents	Section 1.10 Related Documents
Finding I, Criterion 5 – The PA identifies procedures and facility related documentation (e.g., Safety Analysis Report, Operational Readiness Review, Waste Acceptance Criteria) that may impact site engineering, facility design, or facility operations. The PA also describes the impacts of procedures and documents that may be precipitated by the PA results.	Section B.2.4 Related Documents	Section 1.10 Related Documents
Finding I, Criterion 1 – The PA identifies the performance assessment measures and a justification for their use to achieve site-specific applications of the performance objectives.	Section B.2.5 Performance Criteria	Section 1.5 Performance Objectives
Finding I, Criterion 7 – The PA identifies the point of assessment for each performance measure, and justifies the selection of each point of assessment. The point of assessment is the location for which compliance with the performance objectives is evaluated.	Section B.2.5 Performance Criteria	Section 1.5 Performance Objectives
Finding I, Criterion 7a – The point of assessment for all pathways, the air pathway excluding radon, and groundwater resource protection is justified based on future land use. If the future site boundary is uncertain, a reasonable point of assessment (e.g., point of maximum impact greater than 100 m from the edge of the disposal unit) is justified.	Section B.2.5 Performance Criteria	Section 1.9 Scenarios
Finding I, Criterion 7b – The default point of assessment for the performance measure for radon exposure that is based on a limit on the average flux of radon of 20 pCi/m^2/s at the ground surface is the ground surface over the disposal unit.	Section B.2.5 Performance Criteria	Section 1.9 Scenarios
Finding I, Criterion 7c – The default point of assessment for the alternative performance measure for radon exposure that is based on a limit on air concentration of radioactive material of 0.5 pCi/L is 100 m from the edge of the disposal unit.	Section B.2.5 Performance Criteria	Section 6.5 Comparison of Effects of Releases to Air to Performance Objectives
Finding I, Criterion 6 – The PA identifies and justifies the key assumptions included in the analysis.	Section B.2.6 Summary of Key Assessment Assumptions	Section 3.4 Values and Assumptions

Table A-1. Crosswalk of *Low-Level Waste Disposal Facility Federal Review Group Manual* Criteria to the
Initial Single-Shell Tank System Performance Assessment for the Hanford Site (14 pages)

LFRG Manual Criteria [a]	Content Guide Section and Title [b]	Location in SST PA
Finding I, Criterion 13 – The PA provides a coherent presentation of the relevant descriptive information concerning the site, the disposal facility, the waste characteristics that are reflected in the conceptual model, and the selection of the mathematical models used in the analysis. The descriptive information and the approach to modeling provide the necessary results to evaluate the exposure pathways and scenarios that are important to assess the performance of the disposal facility.	Section B.3 Disposal Facility Characteristics	Chapter 2.0 Site and Waste Characteristics Chapter 3.0 Analysis of Performance
Finding I, Criterion 2 – The PA presents information on the following that is sufficient to support the analysis presented in the PA: site geography, demography, land use plans, meteorology, ecology, geology, seismology, volcanology, surface water and groundwater hydrology, geochemistry, geologic resources, water resources, and natural background radiation.	Section B.3.1 Site Characteristics	Section 2.2 Overview Section 2.3 Hanford Site Characteristics
Finding I, Criterion 3 – The PA presents information on the facility design features that address water infiltration, disposal unit cover integrity, structural stability, and the inadvertent intruder barrier sufficient to support the analysis presented in the PA.	Section B.3.2 Principal Facility Design Features	Section 1.6 Defense in Depth Philosophy Applied to Tank Farm Closure Section 1.7 Planned Single-Shell Tank System Closure Actions and End State Section 2.4 Facility Descriptions Common to All Tank Farms Section 3.4.2 Surface Barrier and Pre- and Post-Barrier Recharge Rates
Finding I, Criterion 8 – The PA identifies and quantifies all radionuclides present in the low-level waste to be disposed of at the facility that could significantly contribute to dose for the all pathways analysis, the air pathway analysis, the groundwater analysis, and the intruder analysis. Technical justification is provided for those radionuclides considered in detail in the analysis, and conversely, those not considered in the analysis.	Section B.3.3 Waste Characteristics	Section 2.5 Source Term Inventory Appendix C Inventory Inputs to Performance Assessment Modeling
Finding II, Criterion 1 – The PA presents an estimate of the radionuclide inventory of the radioactive waste disposal of and to be disposed of at the facility which is quantified and technically supported by records, data, studies, and evaluations.	Section B.3.3 Waste Characteristics	Section 2.5 Source Term Inventory Appendix C Inventory Inputs to Performance Assessment Modeling

Table A-1. Crosswalk of *Low-Level Waste Disposal Facility Federal Review Group Manual* Criteria to the *Initial Single-Shell Tank System Performance Assessment for the Hanford Site* (14 pages)

LFRG Manual Criteria [a]	Content Guide Section and Title [b]	Location in SST PA
Finding II, Criterion 1a – All of the radionuclides disposed and anticipated to be present in wastes to be disposed of are evaluated in the PA. Radionuclides screened from detailed analysis or having no inventory limit are identified, and the bases for these conclusions are supported and defensible.	Section B.3.3 Waste Characteristics	Section 2.5 Source Term Inventory Appendix C Inventory Inputs to Performance Assessment Modeling
Finding II, Criterion 1b – Estimates of the radionuclide inventory for past waste disposals are described and to the extent practical, are based on past waste disposal records, a reasonable expectation of actual waste content based on a knowledge of the processes that generated the waste, calculations, sampling data, technical studies, and reasonable projections of waste to be disposed of.	Section B.3.3 Waste Characteristics	Section 2.5 Source Term Inventory Appendix C Inventory Inputs to Performance Assessment Modeling
Finding II, Criterion 2 – The physical and chemical characteristics of the waste disposed of in the past that affect the release and transport of radionuclides are identified. The physical and chemical characteristics of the waste form are quantified and supported by laboratory or field studies, or are based on referenced documentation.	Section B.3.3 Waste Characteristics	Section 2.5 Source Term Inventory Appendix C Inventory Inputs to Performance Assessment Modeling
Finding II, Criterion 3 – Any inventory limits are developed from reasonable projections of waste to be disposed and analyses that consider the physical and chemical characteristics of the wastes if those characteristics affect the release and transport of radionuclides.	Section B.3.3 Waste Characteristics	Section 2.5 Source Term Inventory Appendix C Inventory Inputs to Performance Assessment Modeling
Finding I, Criterion 13 – The PA provides a coherent presentation of the relevant descriptive information concerning the site, the disposal facility, the waste characteristics that are reflected in the conceptual model, and the selection of the mathematical models used in the analysis. The descriptive information and the approach to modeling provide the necessary results to evaluate the exposure pathways and scenarios that are important to assess the performance of the disposal facility.	Section B.4 Analysis of Performance	Chapter 3.0 Analysis of Performance

Table A-1. Crosswalk of *Low-Level Waste Disposal Facility Federal Review Group Manual* Criteria to the *Initial Single-Shell Tank System Performance Assessment for the Hanford Site* (14 pages)

LFRG Manual Criteria [a]	Content Guide Section and Title [b]	Location in SST PA
Finding II, Criterion 4 – The conceptual model is a reasonable interpretation of the existing geochemical, geographic, meteorologic, hydrologic, and monitoring data for the site and disposal facility. The components of the conceptual model for the transport of radionuclides that are important to the conclusions relating to the long-term performance of the disposal facility are thoroughly analyzed. The assumptions incorporated into the conceptual model are consistent with the available data, related investigations, and theory related to the conceptual model. Parameters included in the conceptual model are supported by data or related investigations relevant to the site and disposal facility.	Section B.4.2 Conceptual Model of Facility Performance	Section 1.6 Defense in Depth Philosophy Applied to Tank Farm Closure Section 1.7 Planned Single-Shell Tank System Closure Actions and End State Section 3.2 Performance Assessment Methodology
Finding II, Criterion 5 – The assumptions of the PA related to the waste, site, and facility design and operations which are critical to the conclusions of the PA are supported and the uncertainties associated with these assumptions are analyzed as part of the PA. Credits for the performance of engineered features and site closure included in the conceptual model are based on data derived from field investigations, related investigations, or documented sources of information relevant to the site and disposal facility.	Section B.4.2 Conceptual Model of Facility Performance	Section 3.2 Performance Assessment Methodology Section 3.4 Values and Assumptions
Finding I, Criteria 9 – The PA accounts for all relevant mechanisms for the release of radionuclides from the waste materials for environmental transport. The mechanisms that are analyzed are justified by references to relevant studies, available data, or supporting analyses in the PA.	Section B.4.2.1 Source Term	Section 3.4 Values and Assumptions
Finding II, Criterion 6 – The conceptual model for the source term, groundwater flow, and radionuclide transport includes parameters for unsaturated and saturated flow, total and effective porosity, hydraulic conductivity, water retention, relative permeability relationships, volumetric water content, retardation, and diffusion that are based on data, related investigations, or documented references relevant to the site and disposal facility.	Section B.4.2.1 Source Term	Section 3.4 Values and Assumptions
Finding I, Criteria 10 – The PA provides a complete and clear description of the conceptual model of the environmental transport of radionuclides from the waste materials to the points of compliance by air and water. The conceptual model is justified by referenced investigations, data, and supporting analyses that are representative of the site-specific conditions described.	Section B.4.2.2 Radionuclide Transport	Section 3.2 Performance Assessment Methodology

Table A-1. Crosswalk of *Low-Level Waste Disposal Facility Federal Review Group Manual* Criteria to the *Initial Single-Shell Tank System Performance Assessment for the Hanford Site* (14 pages)

LFRG Manual Criteria [a]	Content Guide Section and Title [b]	Location in SST PA
Finding I, Criterion 10a – The conceptual model incorporates interpretations of available geochemical, geologic, meteorologic and hydrologic data, and the relevant mechanisms that have a significant effect on the transport of radionuclides at the disposal site.	Section B.4.2.2 Radionuclide Transport	Section 3.2 Performance Assessment Methodology
Finding I, Criterion 10b – Assumptions incorporated into the conceptual model to account for transport mechanisms lacking sufficient data or supporting analyses are identified and justified as reasonable representations of site behavior over the time period considered in the analysis.	Section B.4.2.2 Radionuclide Transport	Section 3.2 Performance Assessment Methodology
Finding I, Criterion 10c – The conceptual model includes closure of the facility as justified based on referenced closure plans or reasonable assumptions of facility closure.	Section B.4.2.2 Radionuclide Transport	Section 1.7 Planned Single-Shell Tank System Closure Actions and End State Section 3.2 Performance Assessment Methodology
Finding I, Criterion 10d – The conceptual model includes any credits to be taken in the analysis for the performance of engineered features. Credits for engineered features include a reasonable representation of the degradation of the engineered features that is justified by supporting investigations and data.	Section B.4.2.2 Radionuclide Transport	Section 3.4 Values and Assumptions
Finding I, Criterion 10e – The conceptual model includes natural processes that affect the transport of radionuclides (e.g., flooding, mass wasting, erosion, weathering) over the time period considered in the analysis, as justified based on referenced investigations and supporting analysis.	Section B.4.2.2 Radionuclide Transport	Section 3.4 Values and Assumptions
Finding II, Criterion 6 – The conceptual model for the source term, groundwater flow, and radionuclide transport includes parameters for unsaturated and saturated flow, total and effective porosity, hydraulic conductivity, water retention, relative permeability relationships, volumetric water content, retardation, and diffusion that are based on data, related investigations, or documented references relevant to the site and disposal facility.	Section B.4.2.2 Radionuclide Transport	Section 3.4 Values and Assumptions

Table A-1. Crosswalk of *Low-Level Waste Disposal Facility Federal Review Group Manual* Criteria to the
Initial *Single-Shell Tank System Performance Assessment for the Hanford Site* (14 pages)

LFRG Manual Criteria [a]	Content Guide Section and Title [b]	Location in SST PA
Finding I, Criterion 12 – The PA provides a complete description of the important exposure pathways and scenarios for the specific disposal facility that are used in the evaluation of the potential doses to a hypothetical, individual member of the public and inadvertent intruder consistent with site-specific environmental conditions and local and regional practices. The exposure pathways and scenarios selected for detailed analysis are justified as conservative representations of the long-term performance of the LLW disposal facility. These include:	Section 4.2.3 Exposure Pathways and Scenarios	Section 1.9 Scenarios Section 3.4 Values and Assumptions
Finding I, Criterion 12a – Exposure pathways from the transport of contamination in groundwater that may be considered include potential exposures from the ingestion of contaminated groundwater, the use of contaminated groundwater for irrigation and livestock watering, and the biotic uptake and transport of contamination from groundwater to surface water. Potential exposure pathways from the transport of contamination in surface water include the ingestion of contaminated surface water and contaminated fish.	Section 4.2.3 Exposure Pathways and Scenarios	Section 1.9 Scenarios Section 3.4 Values and Assumptions
Finding I, Criterion 12b – If radiation dose is used as a measure of groundwater resource protection, the exposure scenarios consider the ingestion of water (at 2 liters per day or an alternative rate, if a justification is included) at the point of assessment, which represents the location of maximum exposure from a well constructed and developed using current practices typical for the local area.	Section 4.2.3 Exposure Pathways and Scenarios	Section 1.9 Scenarios Section 3.4 Values and Assumptions
Finding I, Criterion 12c – Exposure scenarios from the transport of contamination in water for the all pathways analysis considers the use of groundwater and surface water consistent with local and regional practices. Exposure scenarios that may be considered include drinking water, crop irrigation and livestock watering, the ingestion of dairy products, livestock, fish, crops, and soil, the inhalation of resuspended particles, and external exposure.	Section 4.2.3 Exposure Pathways and Scenarios	Section 1.9 Scenarios Section 3.4 Values and Assumptions
Finding I, Criterion 12d – Exposure pathways from the transport of contamination in the atmosphere that may be considered include potential exposure from immersion in air contaminated with volatile and nonvolatile radionuclides, deposition of volatile and nonvolatile radionuclides, and subsequent exposure from direct radiation, ingestion, and resuspension.	Section 4.2.3 Exposure Pathways and Scenarios	Section 1.9 Scenarios Section 3.4 Values and Assumptions

Table A-1. Crosswalk of *Low-Level Waste Disposal Facility Federal Review Group Manual* Criteria to the *Initial Single-Shell Tank System Performance Assessment for the Hanford Site* (14 pages)

LFRG Manual Criteria [a]	Content Guide Section and Title [b]	Location in SST PA
Finding I, Criterion 12e – Exposure scenarios from the transport of contamination in air that may be considered include residential and gardening activities which include the direct inhalation of volatile radionuclides, external exposures, ingestion of crops, soil, livestock, dairy products, and inhalation of resuspended particles.	Section 4.2.3 Exposure Pathways and Scenarios	Section 2.3 Hanford Site Characteristics Section 6.5 Comparison of Effects of Releases to Air to Performance Objectives
Finding I, Criterion 11 – The PA provides a clear description of the mathematical models used in the analysis, the basis for their selection, and their linkage. The mathematical models selected are justified and provide a reasonable representation of all of the elements of the conceptual model.	Section B.4.3 Source Term	Section 3.3 Numerical Implementation
Finding I, Criterion 11a – The complexity of the mathematical models selected is commensurate with the available site data.	Section B.4.3 Source Term	Section 3.3 Numerical Implementation
Finding I, Criterion 11b – Assumptions incorporated into the mathematical models are identified, justified, and consistent with the conceptual model.	Section B.4.3 Source Term	Section 3.3 Numerical Implementation
Finding I, Criterion 11c – Mathematical models selected are documented and verified either in referenced publications or in the appendices of the PA.	Section B.4.3 Source Term	Section 3.3 Numerical Implementation
Finding I, Criterion 11 – The PA provides a clear description of the mathematical models used in the analysis, the basis for their selection, and their linkage. The mathematical models selected are justified and provide a reasonable representation of all of the elements of the conceptual model.	Section B.4.4 Environmental Transport of Radionuclides	Section 3.3 Numerical Implementation
Finding I, Criterion 11a – The complexity of the mathematical models selected is commensurate with the available site data.	Section B.4.4 Environmental Transport of Radionuclides	Section 3.3 Numerical Implementation
Finding I, Criterion 11b – Assumptions incorporated into the mathematical models are identified, justified, and consistent with the conceptual model.	Section B.4.4 Environmental Transport of Radionuclides	Section 3.3 Numerical Implementation
Finding I, Criterion 11c – Mathematical models selected are documented and verified either in referenced publications or in the appendices of the PA.	Section B.4.4 Environmental Transport of Radionuclides	Section 3.3 Numerical Implementation

Table A-1. Crosswalk of *Low-Level Waste Disposal Facility Federal Review Group Manual* Criteria to the Initial Single-Shell Tank System Performance Assessment for the Hanford Site (14 pages)

LFRG Manual Criteria [a]	Content Guide Section and Title [b]	Location in SST PA
Finding II, Criterion 7 – The mathematical models used in the PA for analyzing air and water transport of radionuclides are appropriate for the disposal facility and disposal site. The selected models provide a justified representation of the technically important mechanisms identified in the conceptual model, and provide calculated results that are a defensible basis for formulating conclusions.	Section B.4.4 Environmental Transport of Radionuclides	Section 3.3 Numerical Implementation
Finding II, Criterion 7a – The input data for the mathematical models are derived from field data from the site, laboratory data interpreted for field applications, or reference literature sources which are applicable to the site. Assumptions which are used to formulate input data are justified and have a defensible technical basis.	Section B.4.4 Environmental Transport of Radionuclides	Section 3.4 Values and Assumptions
Finding II, Criterion 7b – Intermediate calculations are performed and results are presented that demonstrate, by comparison to site data or related investigations, that the calculations of the mathematical models used in the PA are representative of disposal site and facility behavior for important mechanisms represented in the mathematical models.	Section B.4.4 Environmental Transport of Radionuclides	Section 3.4 Values and Assumptions
Finding II, Criterion 7c – Representations of groundwater well performance (e.g., construction, diameter, yield, depth of penetration, screen length) are reasonable reflections of regional practices and are justified.	Section B.4.4 Environmental Transport of Radionuclides	Section 3.3 Numerical Implementation
Finding II, Criterion 7d – The mathematical models are tested, by comparison to analytical calculations or other models, to demonstrate that the results are consistent with the conceptual model, physical and chemical processes represented in the models, and available site data. The models are evaluated for defensibility and are reasonable representations of the disposal site and facility performance by comparison to available site data, related technical investigations, or referenced documentation or literature.	Section B.4.4 Environmental Transport of Radionuclides	Section 3.4 Values and Assumptions
Finding II, Criterion 7e – The initial conditions, boundary conditions, and the changes of properties with time for the mathematical model are analytically correct (i.e., well posed), and derived from existing site data and information.	Section B.4.4 Environmental Transport of Radionuclides	Section 3.4 Values and Assumptions
Finding II, Criterion 8 – The dose analysis considers the exposure pathways and transfer factors and calculates the maximum dose using acceptable methodologies and parameters.	Section B.4.7 Dose Analysis	Chapter 4.0 Modeling Results for the Groundwater Pathway

Table A-1. Crosswalk of *Low-Level Waste Disposal Facility Federal Review Group Manual* Criteria to the
Initial *Single-Shell Tank System Performance Assessment for the Hanford Site* (14 pages)

LFRG Manual Criteria [a]	Content Guide Section and Title [b]	Location in SST PA
Finding II, Criterion 8a – The dose analysis for exposures to radionuclides identifies the transfer coefficients between media and justifies the parameters used in the analysis with supporting data or references to the literature.	Section B.4.7 Dose Analysis	Chapter 4.0 Modeling Results for the Groundwater Pathway
Finding II, Criterion 8b – The dose analysis specifies the consumption of radioactively contaminated materials for the exposure pathways evaluated, the inhalation rates of contaminated materials, and the external exposure rates and conditions for radioactive materials. The parameters are justified using references to the literature or site-specific investigations.	Section B.4.7 Dose Analysis	Section 3.4 Values and Assumptions
Finding II, Criterion 8c – The dose analysis is conducted using effective dose equivalents in accordance with ICRP-30 (1979) and uses dose conversion factors from recognized published sources.	Section B.4.7 Dose Analysis	Section 3.4 Values and Assumptions
Finding II, Criterion 8d – The maximum dose projected for 1,000 years after facility closure at the point of compliance is used in the analysis for evaluating disposal of LLW or establishing waste acceptance criteria for future disposals.	Section B.4.7 Dose Analysis	Chapter 4.0 Modeling Results for the Groundwater Pathway
Finding I, Criterion 14 – The calculated results presented in the PA are demonstrated to be consistent with the site characteristics, the waste characteristics, and the conceptual model of the facility. The demonstration of consistency is supported by available site monitoring data and supporting field investigations.	Section B.5 Results of Analyses	Chapter 4.0 Modeling Results for the Groundwater Pathway
Finding II, Criterion 9d – The maximum projected dose and time of occurrence are presented in the PA to provide for understanding of the natural system being modeled and the behavior of the model.	Section B.5.5 Dose Analysis	Chapter 4.0 Modeling Results for the Groundwater Pathway
Finding I, Criterion 15 – The models used for calculating the results presented in the PA are analyzed to identify the sensitive parameters of the analysis. The results of the sensitivity analysis are used to evaluate the uncertainty in the calculated results.	Section B.5.6 Sensitivity and Uncertainty Analysis	Section 4.11 Summary of Results of Sensitivity and "What If" Analyses
Finding I, Criterion 16 – The results of the uncertainty analysis are interpreted as they relate to establishing reasonable assurance that the conclusions of the PA are correct.	Section B.5.6 Sensitivity and Uncertainty Analysis	Section 4.11 Summary of Results of Sensitivity and "What If" Analyses Section 5.6 Sensitivity and Uncertainty in the Dose Results Section 7.3 Sensitivity Analysis

**Table A-1. Crosswalk of *Low-Level Waste Disposal Facility Federal Review Group Manual* Criteria to the
*Initial Single-Shell Tank System Performance Assessment for the Hanford Site*** (14 pages)

LFRG Manual Criteria [a]	Content Guide Section and Title [b]	Location in SST PA
Finding I, Criterion 20 – The PA includes an ALARA analysis, and if appropriate, the analytical methods for the ALARA assessment are described.	Section B.5.6 Sensitivity and Uncertainty Analysis	Section 6.6 ALARA Analysis
Finding II, Criterion 9 – The sensitivity and uncertainty analysis considers those parameters and mechanisms that are important to the conclusions relating to the long-term performance of the disposal facility, including radionuclide inventory, radionuclide characteristics, release rates, site and facility characteristics, groundwater flow parameters, site meteorology, and radionuclide transport parameters. Parametric and mechanistic variations analyzed in the uncertainty analysis that are important to the conclusions are justified as reasonable for the site and facility using data or related field investigations.	Section B.5.6 Sensitivity and Uncertainty Analysis	Section 4.11 Summary of Results of Sensitivity and "What If" Analyses Section 7.3 Sensitivity and "What If" Analysis
Finding II, Criterion 9a – The parameters important to the components of the analysis are analyzed to identify the sensitive parameters, and the selection of sensitive parameters is quantitatively justified.	Section B.5.6 Sensitivity and Uncertainty Analysis	Section 4.11 Summary of Results of Sensitivity and "What If" Analyses Section 7.3 Sensitivity and "What If" Analysis
Finding II, Criterion 9b – The sensitive parameters are analyzed for uncertainty in the results of the analysis to provide quantitative bounds for interpreting the results of the analysis.	Section B.5.6 Sensitivity and Uncertainty Analysis	Section 4.11 Summary of Results of Sensitivity and "What If" Analyses Section 7.3 Sensitivity and "What If" Analysis
Finding II, Criterion 9c – The results of the sensitivity analysis are determined using a prescribed methodology that is technically justified. The results of the analysis provide the necessary information to justify the assumptions and conclusions of the PA.	Section B.5.6 Sensitivity and Uncertainty Analysis	Section 4.11 Summary of Results of Sensitivity and "What If" Analyses Section 7.3 Sensitivity and "What If" Analysis
Finding II, Criterion 10 – The ALARA analysis provides a cost-benefit analysis that is an optimization of the collective or population dose based on the cost of dose reduction in the exposed population of $1,000 to $10,000 per person-rem averted. (ALARA analysis is not required if the projected individual or collective doses in the exposed population are trivial.)	Section B.5.6 Sensitivity and Uncertainty Analysis	Section 6.6 ALARA Analysis

Table A-1. Crosswalk of *Low-Level Waste Disposal Facility Federal Review Group Manual* Criteria to the *Initial Single-Shell Tank System Performance Assessment for the Hanford Site* (14 pages)

LFRG Manual Criteria [a]	Content Guide Section and Title [b]	Location in SST PA
Finding I, Criterion 12f – Exposure pathways from inadvertent intrusion into the waste disposal units identify the chronic and acute exposure pathways for each of the exposure scenarios considered. The exposure pathways include all relevant ingestion, external exposure, and inhalation pathways for each exposure scenario. [Direct ingestion of contaminated groundwater and exposures to radon should not be considered for inadvertent intrusion, because they are considered separately.]	Section B.6 Inadvertent Intruder Analysis	Section 1.9 Scenarios Chapter 5.0 Inadvertent Intruder Analysis for Residual Waste
Finding II, Criterion 11 – The inadvertent intruder analysis considers the natural and manmade processes that impact the possible exposure to an intruder and calculates the dose using acceptable methodologies and parameters.	Section B.6 Inadvertent Intruder Analysis	Chapter 5.0 Inadvertent Intruder Analysis for Residual Waste
Finding II, Criterion 11a – The inadvertent intruder analysis specifies the reductions in concentrations of radioactive material from mixing with uncontaminated material or the transport of radionuclides from the disposed waste mass, and justifies the parameters used in the analysis with site data, supporting analysis, or referenced information.	Section B.6 Inadvertent Intruder Analysis	Chapter 5.0 Inadvertent Intruder Analysis for Residual Waste
Finding II, Criterion 11b – The inadvertent intruder analysis accounts for naturally occurring processes (e.g., erosion, precipitation, flooding) and the degradation of engineered barriers in the calculation of results.	Section B.6 Inadvertent Intruder Analysis	Chapter 5.0 Inadvertent Intruder Analysis for Residual Waste
Finding II, Criterion 11c – The inadvertent intruder analysis calculates the maximum dose from disposed materials during the period of 100 – 1,000 years after site closure for waste acceptance criteria for wastes to be disposed of in the disposal facility using the recommendations of ICRP-30 (1979) and dose conversion factors from recognized published sources.	Section B.6 Inadvertent Intruder Analysis	Chapter 5.0 Inadvertent Intruder Analysis for Residual Waste
Finding I, Criterion 12g – Acute exposure scenarios for inadvertent intrusion considers direct intrusion into the disposal site and exhumation of accessible waste material. Relevant scenarios that may be considered include discovery, residential construction, and well drilling that incorporate external exposure, inhalation of resuspended particles, and ingestion of particles.	Section B.6 Inadvertent Intruder Analysis	Section 3.2.4 Intruder Pathway Chapter 5.0 Inadvertent Intruder Analysis for Residual Waste

Table A-1. Crosswalk of *Low-Level Waste Disposal Facility Federal Review Group Manual* Criteria to the
Initial Single-Shell Tank System Performance Assessment for the Hanford Site (14 pages)

LFRG Manual Criteria [a]	Content Guide Section and Title [b]	Location in SST PA
Finding I, Criterion 12h – Chronic exposure scenarios for inadvertent intrusion consider direct intrusion into the disposal site and exhumation of accessible waste material. Relevant scenarios that may be considered include residential use and post-construction, and post-drilling agricultural use that incorporate the ingestion of foodstuffs, ingestion of soil, external exposure, and inhalation of resuspended particles.	Section B.6.2 Chronic Scenarios	Section 1.9 Scenarios Chapter 5.0 Inadvertent Intruder Analysis for Residual Waste
Finding I, Criterion 16 – The results of the uncertainty analysis are interpreted as they relate to establishing reasonable assurance that the conclusions of the PA are correct.	Section B.6.4 Intruder Sensitivity/Uncertainty Analysis	Chapter 5.6 Sensitivity and Uncertainty in the Dose Results
Finding I, Criterion 18 – The PA includes an interpretation of the results that allows for a comparison to the performance measures used in the PA, and include any necessary limitations on facility design or operations that are required to meet the performance objectives.	Section B.8 Performance Evaluation	Chapter 6.0 Performance Evaluation
Finding III, Criterion 1 – The PA presents valid conclusions that demonstrate that the all pathways analysis, air pathway analysis, groundwater resource protection analysis, and inadvertent intruder analysis meet the performance objectives of DOE Order 435.1.	Section B.8 Performance Evaluation	Chapter 6.0 Performance Evaluation
Finding III, Criterion 5 – The analysis, results, and conclusions of the PA provide both a reasonable representation of the disposal facility's long-term performance and a reasonable expectation that the disposal facility will remain in compliance with DOE Order 435.1.	Section B.8 Performance Evaluation	Chapter 6.0 Performance Evaluation
Finding III, Criterion 1a – The all pathways performance objective of 25 mrem/year effective dose equivalent is met over the performance period of 1,000 years for all radionuclides disposed of in the disposal facility.	Section B.8.1 Comparison of Results to Performance Objectives	Section 6.3 Comparison of Estimated Groundwater Pathway Human Health Impacts to Performance Objectives at Waste Management Area Fenceline
Finding III, Criterion 1b – The air pathways performance objective of 10 mrem/year effective dose equivalent is met over the performance period of 1,000 years for all radionuclides disposed of in the disposal facility.	Section B.8.1 Comparison of Results to Performance Objectives	Section 6.5 Comparison of Effects of Releases to Air to Performance Objectives
Finding III, Criterion 1c – The radon performance objective of an average flux of 20 pCi/m²/s at the disposal surface or 0.5 pCi/L in air at the point of compliance is met over the performance period of 1,000 years for all radionuclides disposed of in the disposal facility.	Section B.8.1 Comparison of Results to Performance Objectives	Section 6.5 Comparison of Effects of Releases to Air to Performance Objectives

**Table A-1. Crosswalk of *Low-Level Waste Disposal Facility Federal Review Group Manual* Criteria to the
Initial Single-Shell Tank System Performance Assessment for the Hanford Site (14 pages)**

LFRG Manual Criteria [a]	Content Guide Section and Title [b]	Location in SST PA
Finding III, Criterion 1d – The groundwater resource performance measures for all radionuclides to be disposed of in the disposal facility are met over the performance period of 1,000 years at the prescribed point of compliance.	Section B.8.1 Comparison of Results to Performance Objectives	Section 6.2 Comparison of Estimated Groundwater and Surface Water Concentrations to Groundwater Resources Performance Objectives
Finding III, Criterion 1e – The inadvertent intruder performance objectives of 100 mrem/year effective dose equivalent for chronic exposure and 500 mrem effective dose equivalent for acute exposure are met within the disposal facility over the performance period of 1,000 years.	Section B.8.1 Comparison of Results to Performance Objectives	Section 6.4 Comparison of Inadvertent Human Intruder Impacts to Performance Objectives
Finding III, Criterion 1f – The condition that doses from the disposal of waste are ALARA has been demonstrated and incorporated into the design and operations of the disposal facility.	Section B.8.1 Comparison of Results to Performance Objectives	Chapter 6.6 ALARA Analysis
Finding III, Criterion 3 – The conclusions of the PA are applied to the facility design and operations. The resulting design constraints and limitations on operations can be reasonably accomplished at the disposal facility.	Section B.8.2 Use of Performance Assessment Results	Chapter 7.0 Interpretation of Results
Finding III, Criterion 4 – The conclusions of the PA address and incorporate constraints included in Federal, state, and local statutes or regulations or agreements that impact the site design, facility design, or facility operations. The conclusions also address and incorporate any procedural or site documentation changes or constraints due to the results of the facility PA. Reasonable assurance exists that these constraints and impacts are appropriately addressed in the PA.	Section B.8.2 Use of Performance Assessment Results	Chapter 7.0 Interpretation of Results
Finding I, Criterion 17 – The PA integrates the results of the analysis, the uncertainty analysis, the performance measures, waste acceptance criteria, operating procedures, and applicable laws, regulations, policies and agreements to formulate conclusions.	Section B.7 Interpretation of Results	Chapter 7.0 Interpretation of Results
Finding II, Criterion 12 – The results of the analyses for transport of radionuclides and the inadvertent intrusion into the disposal facility, and the sensitivity and uncertainty of the calculated results are comprehensive representations of the existing knowledge of the site and the disposal facility design and operations.	Section B.7 Interpretation of Results	Chapter 7.0 Interpretation of Results

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Table A-1. Crosswalk of *Low-Level Waste Disposal Facility Federal Review Group Manual* Criteria to the *Initial Single-Shell Tank System Performance Assessment for the Hanford Site* (14 pages)

LFRG Manual Criteria [a]	Content Guide Section and Title [b]	Location in SST PA
Finding III, Criterion 2 – The PA conclusions incorporate the findings of the calculated results for the all pathways analysis, air pathway analysis, groundwater resource protection analysis, inadvertent intruder analysis, and sensitivity and uncertainty analysis. The results are interpreted and integrated to formulate conclusions which are supported by the results and the uncertainties in the results.	Section B.7 Interpretation of Results	Chapter 7.0 Interpretation of Results
Finding I, Criterion 19 – The PA discusses the quality assurance measures applied to the preparation of the analysis and its documentation.	Section B.11 Appendices	Appendix F Quality Assurance
Finding I, Criterion 21 – The PA includes appendices or references to published documents and/or data that provide a basis for the discussions and analysis in the PA.	Section B.10 References Section B.11 Appendices	References Listed in each chapter and appendix Appendices A through F
This section should list the preparers of the performance assessment, including their qualifications.	—	Appendix G Preparers

[a] LFRG Manual, DOE 1999b, *Low-Level Waste Disposal Facility Federal Review Group Manual.*

[b] DOE 1999a, *Format and Content Guide for U.S. Department of Energy Low-Level Waste Disposal Facility Performance Assessments and Composite Analyses.*

ALARA = as low as reasonably achievable
LLW   = low-level waste
PA   = performance assessment
SST   = single-shell tank
SST PA = *Initial Single-Shell Tank System Performance Assessment for the Hanford Site*

A2.0   REFERENCES

DOE O 435.1, 2001. *Radioactive Waste Management*, U.S. Department of Energy, Washington, D.C.

DOE 1999a. *Format and Content Guide for U.S. Department of Energy Low-Level Waste Disposal Facility Performance Assessments and Composite Analyses*, U.S. Department of Energy, Washington, D.C.

DOE 1999b. *Low-Level Waste Disposal Facility Federal Review Group Manual*, Rev. 1, U.S. Department of Energy, Washington, D.C.

ICRP-30 1979. *Limits for the Intake of Radionuclides by Worker*s, ICRP Publication 30, International Commission on Radiological Protection, Stockholm, Sweden.

APPENDIX B

HANFORD SITE PROCESS CHEMISTRY HISTORY

CONTENTS

TERMS

1C   1st cycle waste
2C   2nd cycle waste
DBBP dibutylbutyl phosphonate
ITS   in-tank solidification
LAW   low-activity waste
PIF   Plutonium Isolation Facility
PRF   Plutonium Reclamation Facility
PRTR Plutonium Recycle Test Reactor
PUREX plutonium-uranium extraction
REDOX reduction-oxidation
SIM   Soil Inventory Model
SST   single-shell tank
TBP   tributyl phosphate
UNH   uranium nitrate hexahydrate
UO_3 uranium trioxide
UPR   unplanned release

B1.0 INTRODUCTION

The Hanford Site was established in 1943 as the location for developing plutonium production operations as part of the Manhattan Project. The Manhattan Project had developed two approaches in attempting to create an atomic bomb. Shortly after nuclear fission had been reported by Hahn and Strassmann (1939), scientists recognized the potential for developing a weapon based on uranium-235 fission. However, developing technologies to separate the naturally occurring uranium-235 from the far more abundant uranium-238 was a daunting task; that effort was undertaken at Oak Ridge, Tennessee. The second approach involved producing a new transuranic element through neutron bombardment that would have the nuclear properties required to function in a weapon. The Hanford Site was chosen as the site for this massive industrial undertaking. Since elemental separations based on chemical processes are much more efficient than the isotopic separations required to produce highly enriched uranium-235, the large-scale production of plutonium was a promising approach if the other technical issues related to weapon construction could be resolved. (A very readable discussion of technical issues associated with the development of the Manhattan Project can be found in Richard Rhodes' 1986 book, *The Making of the Atomic Bomb*.)

To support this plutonium production mission at the Hanford Site, initially three nuclear reactors (B, D, and F) were constructed along the Columbia River in the 100 Area. Chemical processing of the irradiated uranium fuel rods was centered in the 200 Areas, and uranium fuel production and research and development activities were centered in the 300 Area, near the town of Richland, Washington (Figure B-1).

There is a direct link between the various chemical separations processes conducted at the Hanford Site and the wastes currently stored in the underground tanks in the 200 Areas and the extensive discharges of radioactive fluids to the soil column. For a number of reasons, essentially all of the very high-activity waste streams generated during plutonium recovery operations at the Hanford Site prior to 1980 have been reprocessed. Often, these high-activity waste streams were reprocessed multiple times by physical, chemical, and thermal means. In many cases, reprocessed high-activity waste streams were commingled with lower activity wastes to produce the materials currently stored in the tanks. The goal of this appendix is to provide a technical overview of the various chemical processing operations in order to provide a basis for understanding the current waste types stored in the single-shell tanks (SST), as well as to provide a connection between the chemical separations processes and the various waste types that were discharged to the environment.

A timeline of the major Hanford Site 200 Areas operations is shown in Figure B-2. Between 1944 and 1964, nine production reactors were constructed along the Columbia River along with a number of test reactors. These reactors provided the vast majority of irradiated fuel reprocessed in the 200 Area facilities. This appendix focuses on the 200 Area chemical reprocessing facilities.

1 Figure B-3 provides a timeline for uranium fuel reprocessing at the Hanford Site. It is interesting
2 to note that approximately 74% of the reprocessing was completed at the Plutonium-Uranium
3 Extraction (PUREX) Plant. Thus, this fraction of total fission products would be associated with
4 PUREX waste streams. As noted in Figure B-3, only about 7% of the fuel was processed using
5 the bismuth phosphate process in B and T Plants. However, as will be discussed in detail further,
6 an inordinate amount of the chemical wastes currently stored in the underground storage tanks
7 were associated with this 7% of reprocessed fuel.

8 **Figure B-1. Hanford Site Map and Location in Washington State**

9
10
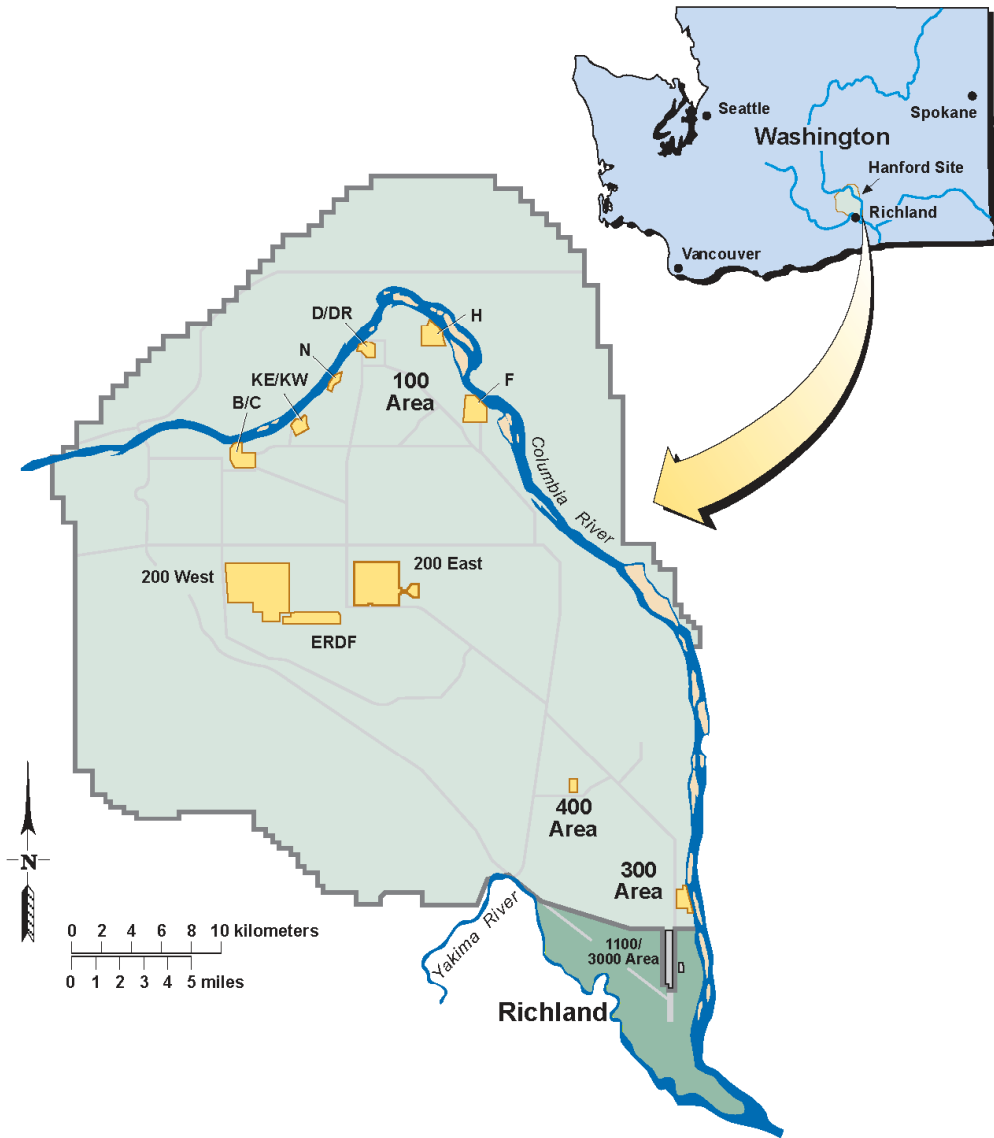
Figure B-2. Timeline of the Major Hanford Site 200 Areas Operations (2 pages)
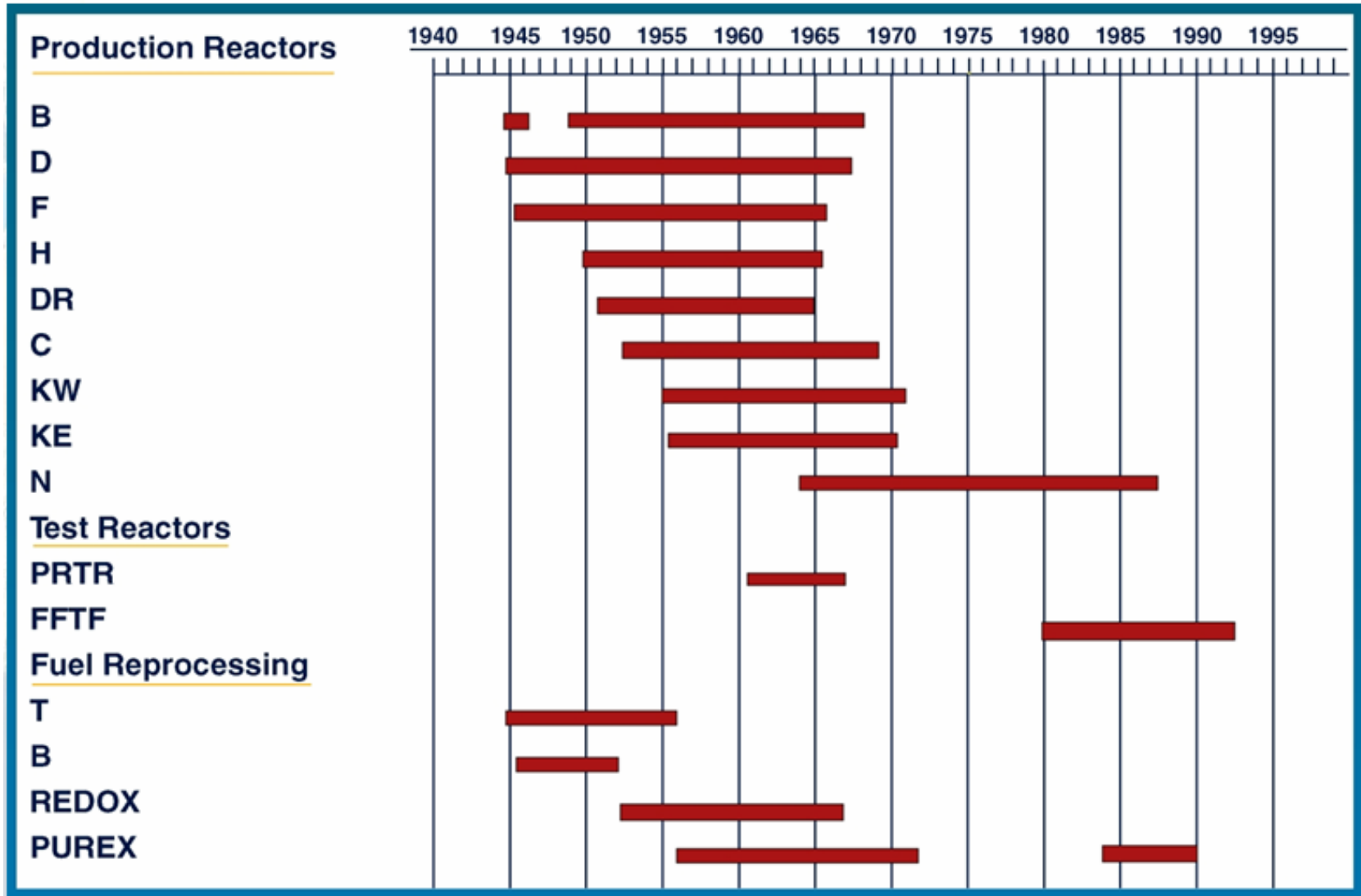
Figure B-2. Timeline of the Major Hanford Site 200 Areas Operations (continued)
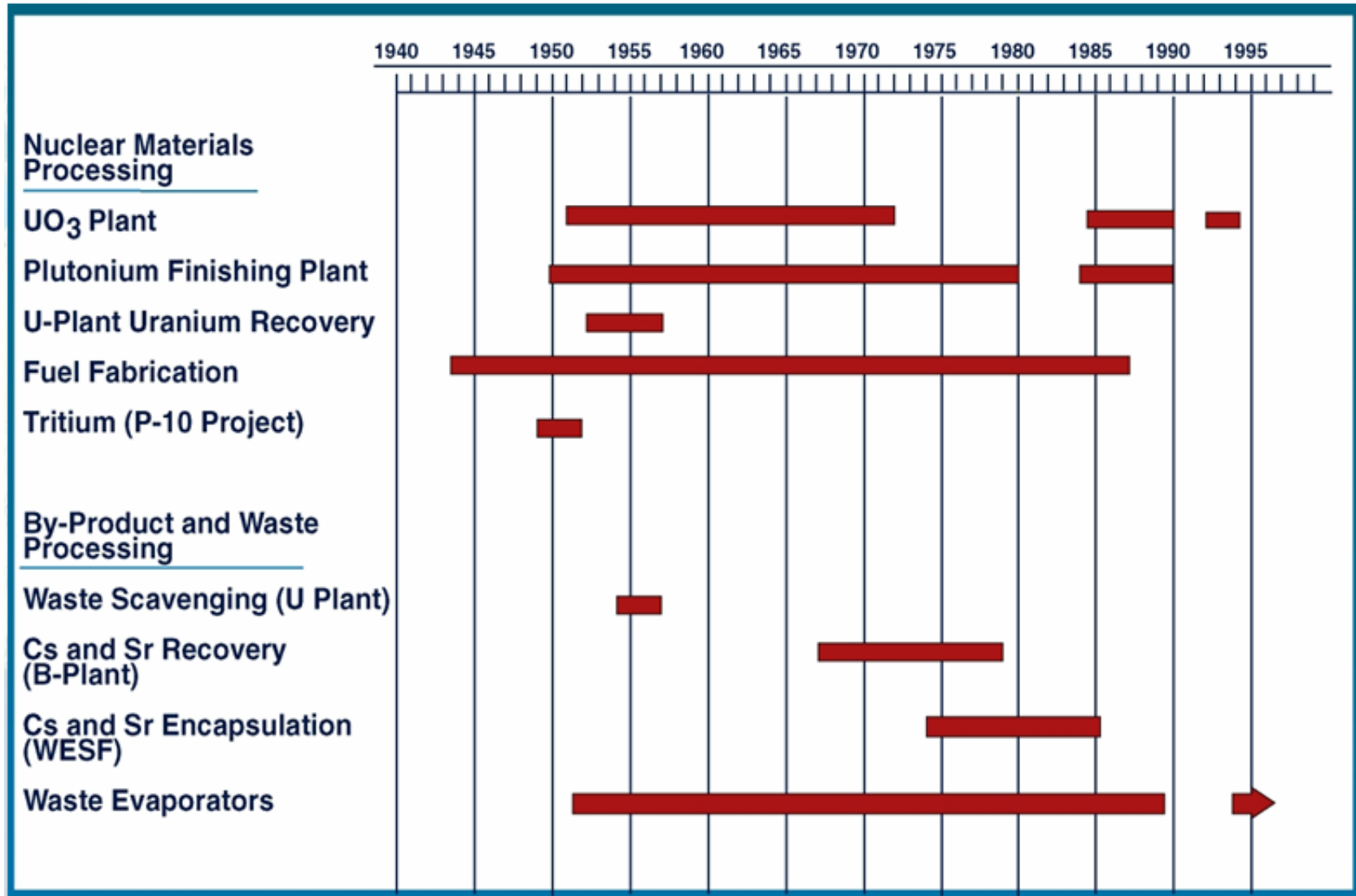
Figure B-3. Timeline for Uranium Fuel Reprocessing at the Hanford Site
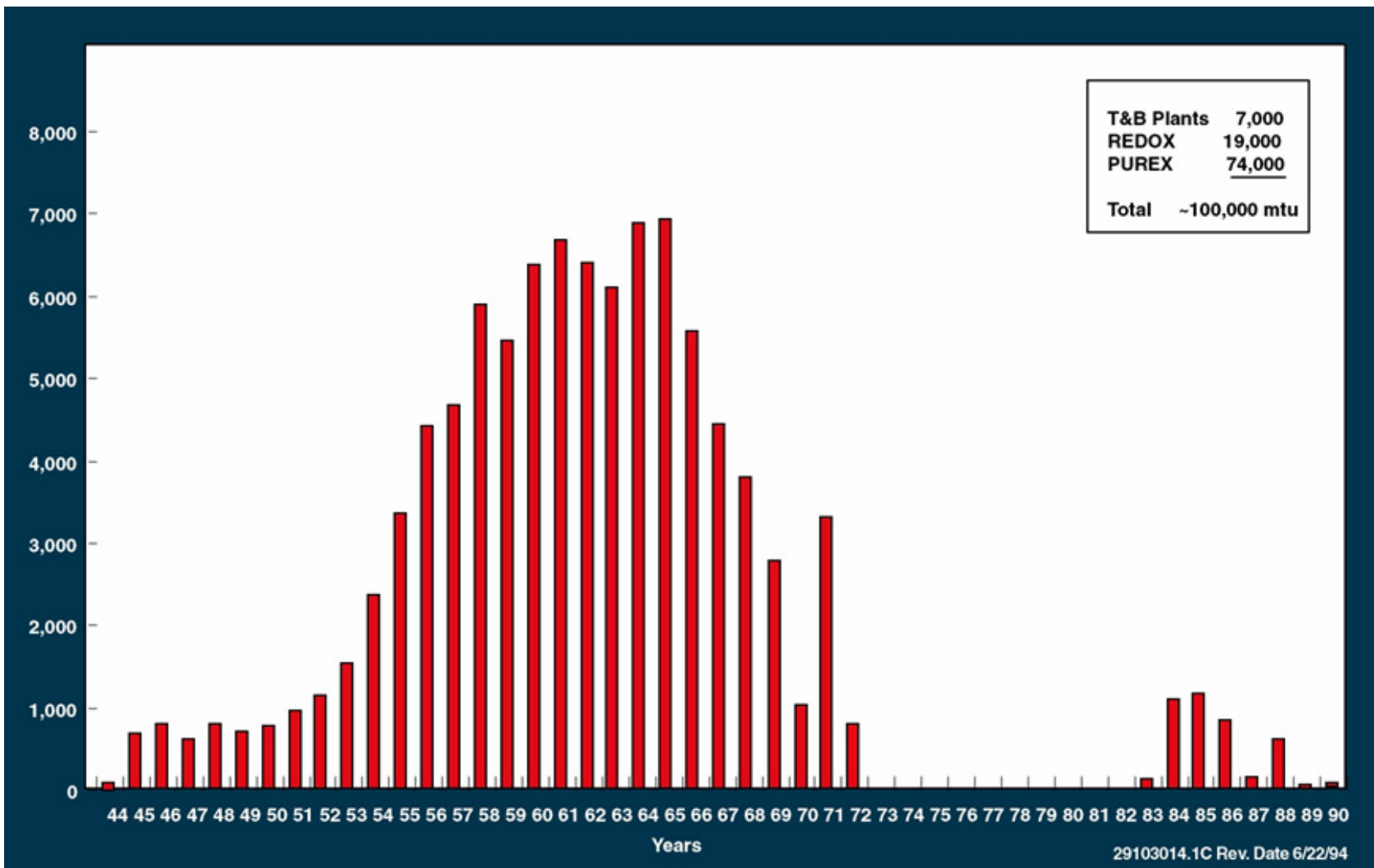
The major chemical separation, purification, and waste treatment operations that were conducted in the 200 Areas can be divided into four areas:

- Plutonium and uranium recovery processes
- Isotope recovery operations
- Product finishing processes (plants)
- Waste volume reduction/thermal processes.

The major Hanford Site plutonium and uranium recovery methods were:

- Bismuth phosphate process
- Uranium recovery from bismuth phosphate metal waste
- Reduction-oxidation (REDOX) process
- PUREX process.

Major isotopic recovery programs, based on chemical separations, included:

- Cesium-137 recovery from REDOX and PUREX high-activity supernates
- Strontium-90 recovery from PUREX high-activity sludge
- Uranium-233 recovery from thorium-232 targets.

Other isotopes of interest included neptunium-237, americium-241, and a number of lanthanides.

The following product finishing operations were conducted at the Hanford Site:

- Plutonium finishing operations
- Uranium trioxide (UO_3) production
- Cesium and strontium encapsulation and storage.

The thermal tank waste volume reduction operations included:

- Standalone evaporator operations
- Radiolitic heat leading to boiling waste tanks
- In-tank heaters in BY farm tanks
- In-plant evaporators used during chemical process operation.

Major facilities associated with 200 Area chemical operations include:

- 221-T and B Plants used plutonium recovery using bismuth phosphate process
- 221-U Plant initially used for training; then, after major modifications, used for uranium recovery from metal waste
- 202-S Plant used for REDOX operations
- 202-A Plant used for PUREX operations
- Hot Semi-Works Facility used for REDOX and PUREX process development and later for isotope recovery processes
- Renovated B Plant for large scale isotope recovery processes
- Z Plant complex used for plutonium finishing activities
- UO_3 Plant used for uranium finishing activities.

1　Facilities supporting tank waste volume reduction operations include:

2　　• 242-T and B Evaporators (1951 – 1956)
3　　• In-Tank Solidification using BY farm tanks (1965 – 1974)
4　　• 242-T, S, and A Evaporators
5　　• REDOX Evaporator
6　　• B Plant Evaporator.

7　Large quantities of plutonium and uranium, along with smaller quantities of other radionuclides,
8　were recovered in the various Hanford Site chemical separations plants. These chemical
9　separations processes were based on a combination of oxidation-reduction, precipitation,
10　complexation, and phase-transfer reactions. The initial plutonium recovery process was based on
11　inorganic precipitation reactions, whereas later processes incorporated large-scale, liquid-liquid
12　extraction technologies.

13　The first plutonium recovery process (i.e., the bismuth phosphate process) was basically an
14　inorganic step-wise precipitation process that only recovered plutonium. The bismuth phosphate
15　process generated extremely large volumes of tank waste and returned the uranium byproduct to
16　the underground single-shell storage tanks. The next plutonium recovery methodology, the
17　REDOX process, employed oxidation state changes for selective phase transfer separations
18　between organic and aqueous phases. It was also the first industrial-scale continuous feed
19　solvent extraction process applied to the separations of inorganic materials. Both plutonium and
20　uranium were recovered in the REDOX process. The PUREX process was modeled after
21　REDOX but utilized a different extractant and diluent. The uranium recovery process was used
22　to recover uranium that had been returned to the SSTs during the bismuth phosphate process
23　operations and was a continuous feed solvent extraction system. The uranium recovery
24　processed was the original application of the solvent extraction chemistry that was used later in
25　PUREX.

26　The multitude of chemical separations processes used at the Hanford Site led to the production of
27　a wide array of chemical waste streams. Figure B-4 shows a schematic diagram of typical
28　reprocessing and waste stream production activities that would have been associated with
29　operations of each major plant. Some waste streams contained large quantities of radionuclides
30　in a neutralized nitric acid-based mixture of supernate and sludge. Other waste streams were
31　large volume aqueous streams that contained low levels of radionuclides, and very large volumes
32　of non-contaminated cooling water. Management of the various chemical waste streams
33　generated in the Hanford Site processing plants has always been a daunting challenge. A list of
34　the major waste streams initially routed to the underground storage tanks is shown in Table B-1.

Figure B-4. Overview of Waste Streams and Discharges from Hanford Site Chemical Separation Processes
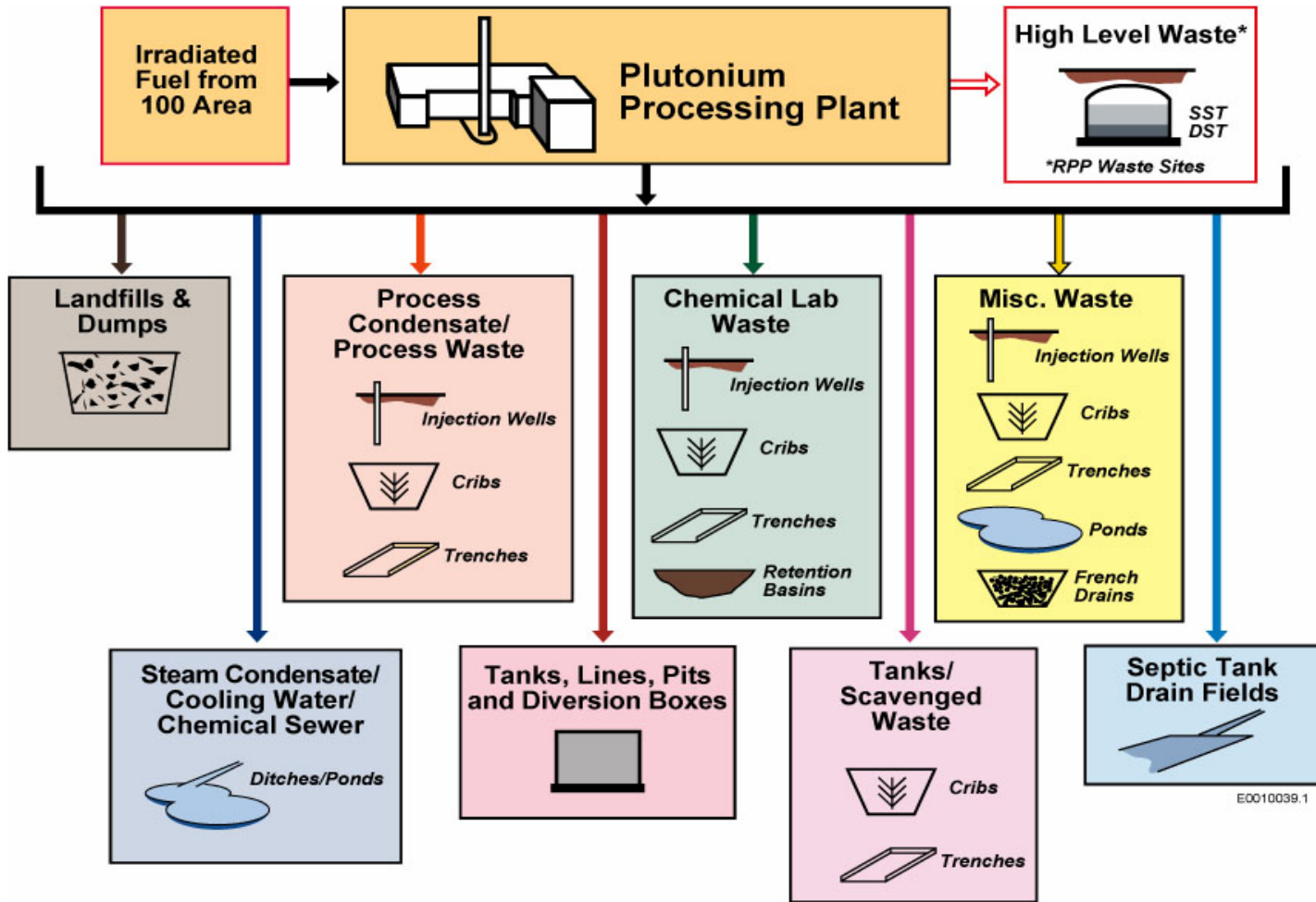
Table B-1. Major Hanford Chemical Processes Waste Streams Going to Underground Storage Tanks

B Plant - Bismuth Phosphate Process (1945 – 1956)
Primary Waste Streams that Went to Tanks
Metal waste (high fission product waste)
First cycle (intermediate radionuclide concentration waste-included aluminum-cladding wastes)
Second cycle (low radionuclide concentration waste)
224 wastes (low radionuclide concentration waste)
REDOX (1952 – 1966)
Primary Waste Streams that Went to Tanks
REDOX high-activity waste
Aluminum cladding waste
Hot Semi-Works (1948 – 1968)
Primary Waste Streams
High-activity waste streams went to C farm tanks, low-activity waste to cribs
Uranium Recovery (1952 – 1958)
Primary Waste Streams
Uranium recovery
Ferrocyanide-scavenged uranium recovery supernate (mostly went to the soil column) Ferrocyanide-scavenged uranium recovery sludges remain in tanks
PUREX (1956 – 1972)
Primary Waste Streams that Went to Tanks
Aluminum-cladding wastes
Zirconium-cladding wastes
Neutralized fission-product wastes to boiling waste tanks
Organic wash wastes
Current acid wastes transferred to B Plant for isotope recovery before going to tanks
B Plant Isotope Recovery Program (1967 – 1978)
Primary B Plant Waste Streams Going to Tanks
Current acid wastes processing waste stream (went to boiling waste tanks)
Cesium-recovery (ion exchange) wastes from aged PUREX supernate
Cesium-recovery (ion exchange) wastes from aged PUREX sludge
Cesium-recovery (ion exchange) wastes from aged REDOX supernate
Strontium-recovery (extraction) wastes from PUREX acidified sludges
Current acid wastes processing waste stream (went to boiling waste tanks)

PUREX = Plutonium-Uranium Extraction
REDOX = Reduction-Oxidation (facility)

1   Due to a critical shortage of tank space in the mid 1950s, approximately 23 million gal of higher
2   activity tank waste were routed to cribs, and approximately 27 million gal of higher activity tank
3   wastes were routed to "specific retention trenches" for disposal. These trenches were engineered
4   to trap contaminates within the soil column well above the groundwater table.

5   Many of the cribs and specific retention trenches are located in the vicinity of SST farms.
6   Therefore, discharges to nearby cribs or specific retention trenches would have contributed
7   vadose zone contamination near these tank farms. It is likely that essentially all of the large
8   volume discharges to cribs would have impacted groundwater quality. Thus, a realistic
9   assessment of the impact of past and future tank farm operations on the environment must
10   include information on known discharges to nearby cribs and specific trenches and other soil
11   disposal sites.

12   Large volumes of non-contaminated water (approximately 134 billion gal) were discharged to
13   surface impoundments or ponds. The discharges to these surface ponds significantly impacted
14   water table levels in the 200 East and West Areas.

B2.0   BISMUTH PHOSPHATE PROCESS (1944 TO 1956)

The T Plant began operations in December 1944 and B Plant began operations in April 1945.
B Plant operated until 1952 and T Plant until 1956. These plants were composed of several
buildings, including the 221-T and 221-B Buildings (also known as the "Canyon Buildings" due
to their shape and appearance) and the 224-T and B Buildings (where the final plutonium
purification procedures were performed). Other facilities include:

- 241-T and 241-B waste disposal buildings
- 222-T and 222-B analytical chemistry laboratories
- 211-T and 211-B tank farms or aboveground chemical storage buildings
- 291-T and 291-B fan houses and process stacks
- 292-T and 292-B stack monitoring buildings
- 271-T and 271-B service buildings.

Initially, four tank farms were constructed to support four plants operating the bismuth phosphate
process, B and C Plants in the 200 East Area, and T and U Plants in the 200 West Area.
Three facilities were constructed (B, T, and U Plants) but only two (B and T Plants) were used to
process irradiated fuel. U Plant was initially used for training and was later modified to support
uranium recovery operations. Initially, B Plant transferred waste to both the B and C tank farms
and T Plant transferred wastes to the T and U tank farms. In the late 1940s, four additional tank
farms were constructed to support the two bismuth phosphate process operations. BX and BY
tank farms were constructed in the 200 East Area near the B tank farm, and TX and TY tank
farms were constructed in the 200 West Area near the T tank farm.

In the late 1940s, a modified C Plant or Hot Semi-Works Facility was constructed at the original
C Plant site for use as a pilot plant and chemical engineering laboratory. The Hot Semi-Works
Facility was used to support the development of solvent-extraction based plutonium recovery
processes that replaced the bismuth phosphate process in the 1950s and in the development of
isotope recovery processes. Operations of the Hot Semi-Works Facility will be discussed along
with those chemical separations processes.

B2.1   BISMUTH PHOSPHATE SEPARATION PROCESS

The uranium rods were fabricated in the 300 Area then transferred to the reactors in the
100 Area. After irradiation, the uranium rods were transferred from the 100 Area to the
200 North Area via shielded railcar. The rods were stored for a short period (generally 45 to
60 days) in large tanks or basins containing water to allow decay of short-lived radionuclides.
After the period of interim storage, the rods were sent via railcar to the 221-T and 221-B
Buildings for processing. The bismuth phosphate process including detailed separations
chemistry is well documented in the *Hanford Works Technical Manual* (GE 1944) and in
updated process flowsheets (Schneider 1951). Overviews of the bismuth phosphate process are
found in Anderson (1990) and Agnew (1997).

The recovery of plutonium by the bismuth phosphate precipitation process involved the selective
adjustment of uranium and plutonium oxidation states in such a manner that plutonium would be
precipitated while uranium remained in solution. In the +4 oxidation state, plutonium would
quantitatively co-precipitate with bismuth phosphate while the uranium in a +6 oxidation state

1   remained in solution. Thus, the first step of the recovery process was to reduce the plutonium to
2   a +4 oxidation state and co-precipitate it with bismuth phosphate. The liquid phase was
3   neutralized and sent to underground storage tanks. Because this liquid phase contained
4   essentially all of the uranium, it was carefully segregated from other waste types and later
5   retrieved from the tanks for recovery of the uranium. This waste stream was labeled "metal
6   waste" because "metal" was the code name for uranium. The metal waste stream also contained
7   essentially all of the water soluble fission and activation products. However, the resulting
8   bismuth phosphate/plutonium(IV) precipitate (i.e., product cake) contained approximately 10%
9   of the total fission product activity. This fission product activity was primarily due to the
10   co-precipitation of zirconium-93/niobium-93 along with low levels of other radionuclides in the
11   plutonium-bismuth phosphate product cake. Given that the goal was a plutonium product that
12   could be handled outside of a shielded facility, additional purification steps were required.
13   Multiple additional dissolution/precipitation steps were required to produce a plutonium product
14   that could be handled outside the shielded facility.

15   The extensive reprocessing, which required using the bismuth phosphate process to obtain the
16   required 10^7 decontamination factor for the plutonium product, led to large volumes of liquid
17   wastes being transferred to the underground storage tanks. Approximately 4,000 gal of metal
18   waste was produced per metric ton of uranium reprocessed. Each dissolution/reprecipitation step
19   generated another 3,000 gal per metric tons of uranium reprocessed. Although the waste streams
20   from the first and second recycle steps were relatively low-activity waste, it was initially stored
21   in the underground tank storage system. Resolution of the waste management issues raised by
22   the large volumes of liquid waste generated during the 12 years of bismuth phosphate process
23   operations led to many of the current conditions in and around the 200 Areas that must be
24   addressed during site closure.

25   **B2.1.1   Fuel Decladding**

26   The uranium rods were encased in an aluminum cladding as a protective jacket prior to being
27   placed in the reactor. The removal of the aluminum cladding was the first step in the fuel
28   reprocessing operations. Two approaches were developed for cladding dissolution: one
29   involving the mercury(II) catalyzed nitric acid dissolution of the aluminum cladding and a
30   second approach used a hot sodium hydroxide/nitrate solution. All available information
31   indicates that the only decladding process used at the Hanford Site involved the dissolution of
32   the aluminum cladding with a boiling hydroxide/nitrate solution. The hot hydroxide/nitrate
33   process was used throughout the duration of site operations when processing aluminum-clad fuel
34   rods. Approximately 500 gal of cladding waste were produced per metric ton of uranium
35   processed. The nitrate ion functioned as an oxidizing agent, producing aluminum(III) ions,
36   nitrite, and a mixture of various nitrogen oxides (NO_X). The aluminum cladding waste consisted
37   of approximately 1.0 M hydroxide, 1.2 M sodium aluminate, 0.6 M nitrate, and 0.9 M nitrite
38   (Anderson 1990).

39   Initially, the uranium rods were coated with a bronze layer prior to being clad in the aluminum
40   jacket. Around 1952, this process was changed and the uranium rods were dipped in molten lead
41   prior to being clad. The uranium rods were bonded to the aluminum cladding with a molten
42   aluminum-silicon. Thus, the cladding waste stream would also include these materials.
43   The levels of radionuclide contamination in the cladding waste stream were dependent on the

1 extent of uranium fuel dissolution taking place during the decladding process. Although it is
2 suggested in GE (1944) that less than 0.1% of the fuel would dissolve during the decladding
3 process, Anderson (1990) estimates that approximately 0.4% of the uranium and plutonium were
4 lost in this step. Although historical documents (e.g., Schneider and Zimmer 1962) indicate that
5 the amounts of uranium and plutonium lost in the cladding waste stream were closely monitored
6 for accountability purposes, the detailed records are currently unavailable.

7 During the bismuth phosphate process, the aluminum cladding waste stream was commingled in
8 the underground storage tanks with another lower activity waste stream from the first recycle
9 step called "1C" or "1st Cycle" waste. Because the cladding waste stream was commingled with
10 the 1C waste stream, there are no specific waste management issues associated with the cladding
11 waste stream produced during the bismuth phosphate process operations. However, the
12 management of aluminum cladding waste did become a waste management issue during the
13 REDOX and PUREX plant operations because this waste consumed large volumes of available
14 tank space.

B2.1.2 Uranium Dissolution

16 The uranium rods were dissolved in approximately 1-metric ton batches in concentrated nitric
17 acid, and digested at the boiling point until a specific gravity of $1.80 – 1.82$ was reached.
18 The dissolver solution was then diluted to approximately 55% uranium nitrate hexahydrate
19 (UNH) to avoid crystallization upon cooling. Sulfuric acid was then added to convert and
20 "stabilize" the plutonium in lower oxidation states (III, IV) and to maintain the uranium(VI) in
21 solution. After the addition of sulfuric acid, the dissolver solution was transferred for further
22 plutonium recovery operations (GE 1944). Along with the uranium and a small quantity of
23 plutonium, this solution contained almost all of the fission and activation products. However,
24 losses of volatile radionuclides occurred during the dissolution process.

25 The uranium dissolution process produced significant quantities (approximately 850 lb per
26 metric ton of uranium processed) of NO_X that were sent to the 291 Building stack for dilution
27 and discharge to the atmosphere. The major volatile radionuclides lost through the air pathway
28 were iodine-129/131 and xenon-133. There were sufficient quantities of iodine-131 in the 200-ft
29 stack effluent to restrict dissolver operations if the weather conditions were unfavorable
30 (e.g., no wind, downdrafts, or fog). During uranium rod dissolution, the off-gas was
31 approximately 100 ft^3/min. Although this off-gas was diluted with 60,000 ft^3/min ventilation
32 air at the base of the stack, a further dilution factor of 500 upon emission between the stack and
33 ground was required before dissolver operations were allowed to proceed.

B2.1.3 Initial Plutonium Precipitation

35 The separation of plutonium from the dissolver solution followed a classic gravimetric
36 precipitation process. The first step involved the addition of sodium nitrite to ensure that
37 essentially all of the plutonium was in a +4 oxidation state. Nitrous acid will quantitatively
38 reduce plutonium(VI) and oxidized plutonium(III) to plutonium(IV) but will not reduce
39 uranium(VI). The next step was the addition of bismuth(III) nitrate followed by slow addition of
40 phosphoric acid. After suitable digestion, the bismuth phosphate product cake that carried the
41 plutonium(IV) was separated by centrifugation. The product cake was then washed with a
42 diluted recycled supernate and then with water. The product cake was primarily bismuth

1 phosphate but also contained co-precipitated plutonium(IV) and approximately 10% of the
2 fission product activity. This activity was primarily the phosphate-insoluble radionuclides
3 zirconium-93 and niobium-93.

4 All liquids from the production of the product cake were combined, neutralized with sodium
5 hydroxide and sodium carbonate to pH 10, and then transferred to underground tanks for interim
6 storage. This waste stream was labeled metal waste and contained essentially all the uranium
7 and fission product mass.

8 Two additional decontamination steps were required before the plutonium product could be
9 removed from the shielded facilities. Each decontamination step required two bismuth
10 phosphate precipitation cycles. The first bismuth phosphate precipitation cycle was carried out
11 with the plutonium in the +6 oxidation state so that it would remain in solution, and the second
12 precipitation cycle with plutonium in the +4 oxidation state so it would co-precipitate with
13 bismuth phosphate.

14 **B2.1.4   First Decontamination (Recycle) Step**

15 The first step of the recycle process was the dissolution of the bismuth phosphate plutonium(IV)
16 product cake with nitric acid. Next, a sodium dichromate solution was added to oxidize the
17 plutonium to a +6 oxidation state, followed by addition of zirconium nitrate, ceric ammonium
18 nitrate, bismuth(III) nitrate, and phosphoric acid. The zirconium and cerium salts were added to
19 enhance the removal of zirconium-93 in the initial precipitation step. After removal of the solids
20 by centrifugation, the plutonium(VI) supernate was treated with sodium nitrite. The addition of
21 sodium nitrite both reduced plutonium(VI) to (IV) and oxidized any plutonium(III) to (IV).
22 The plutonium(IV) was then co-precipitated with bismuth phosphate and the new product cake
23 separated by centrifugation.

24 The product cake was sent to the second decontamination recycle step while all effluents and
25 redissolved waste solids produced in the first recycle step were combined, neutralized to pH 10,
26 and sent to the underground tank storage system. This waste stream is called 1st Cycle or
27 1C waste. Approximately 3,000 gal of 1st Cycle waste were produced per metric ton of uranium
28 processed. As previously noted, the 1C waste was combined with the aluminum cladding in the
29 underground storage tanks.

30 **B2.1.5   Second Recycle Step**

31 The second recycle step closely followed the chemistry used in the first recycle step with the
32 exception that zirconium and cerium salts were not added during the intermediate precipitation
33 step. After dissolution of the product cake with nitric acid, a sodium dichromate solution was
34 added to oxidize the plutonium to a +6 oxidation state, followed by the addition of bismuth(III)
35 nitrate, and phosphoric acid. After removal of the solids by centrifugation, the supernate, which
36 contained the plutonium(VI), was treated with sodium nitrite. The sodium nitrite both reduced
37 plutonium(VI) to (IV) and oxidized any plutonium(III) to (IV). The plutonium(IV) was then
38 co-precipitated with bismuth phosphate and the new product cake separated by centrifugation.
39 At this point, a 3×10^5 decontamination factor had been achieved and an additional
40 decontamination factor of 33 was needed before isolation of the plutonium could occur.

1　This decontamination was obtained in the "Concentrator Building" or 224 Building. The product
2　cake was transferred to the 224 Building for additional plutonium decontamination.

3　The effluents and redissolved waste solids from the second recycle step were combined and
4　neutralized to form the "2nd Cycle" or "2C" waste stream. An estimated 3,000 gal of 2nd Cycle
5　waste was produced per metric ton charge of uranium processed. The 2nd Cycle waste stream
6　was initially routed to the underground storage tanks, but in time, much of the 2nd Cycle
7　supernate was discharged to nearby cribs.

8　**B2.1.6　Lanthanum-Fluoride Decontamination Process**

9　The lanthanum fluoride process was a second part of the bismuth phosphate separation process
10　and purified the plutonium product. It was conducted in the 224-T and 224-B Buildings, or
11　Concentration Building. An understanding of this process is useful because waste streams from
12　the 224 Building were routinely routed to underground tanks where solids settled out and then
13　supernates were routed to the soil column. Although the 224 Building waste streams were very
14　low in beta/gamma activity, these wastes were undoubtedly high in transuranic contamination.

15　The two goals in the 224 Building were to accomplish the additional required decontamination
16　and a reduction in the volume of the plutonium product. These goals were accomplished by
17　switching from bismuth phosphate as the carrier to lanthanum fluoride as the carrier. For this
18　reason, the operations preformed in the 224 Building were commonly referred as the "cross-over
19　cycle."

20　The first step in the cross-over cycle was the dissolution of the bismuth phosphate product
21　cake in nitric acid followed by oxidation of the plutonium(IV) to plutonium(VI). This was
22　followed by a bismuth phosphate precipitation cycle for additional byproduct removal.
23　The plutonium(VI) product remained in the aqueous phase. The next step involved a
24　re-oxidation to ensure the plutonium remained in the +6 oxidation state. The cross-over between
25　carriers occurred here. First, there was a lanthanum fluoride precipitation cycle with plutonium
26　in the +6 oxidation state to remove fluoride-insoluble radionuclides, mainly lantanium-140 and
27　rare earths. After removal of the lanthanum fluoride byproduct solids, the plutonium(VI) was
28　reduced to plutonium(IV) with oxalate. During the second lanthanum fluoride precipitation
29　cycle, the plutonium(IV) co-precipitated with the lanthanum fluoride. The lanthanum fluoride
30　product cake was then treated with potassium hydroxide to convert the fluoride solids into
31　hydroxides. The lanthanum/plutonium hydroxide solids were then dissolved in nitric acid.
32　This lanthanum/plutonium/nitric acid solution (approximately 8 gal) was transferred to the
33　"Isolation Building" (or 231-Z Facility) for final plutonium purification. Approximately 90% of
34　the plutonium product was carried through to this point, and the fission product gamma activity
35　had been reduced by a factor of at least 10^7.

36　**B2.1.7　Final Plutonium Purification**

37　The goal was to produce a plutonium nitrate product that was at least 95% pure, with respect to
38　non-volatile impurities. This was accomplished by carrying out two plutonium peroxide
39　precipitation cycles. The plutonium peroxide precipitation process initially involved the addition
40　of ammonium sulfite to reduce any plutonium(VI) to plutonium(IV) followed by the addition of
41　hydrogen peroxide to form the plutonium peroxide precipitate. The plutonium peroxide

1 precipitate was redissolved in nitric acid and the precipitation process repeated. The plutonium
2 peroxide precipitate was then redissolved in nitric acid and concentrated to form the plutonium
3 nitrate product (generally with 98% purity) that was ready for shipment offsite. Although
4 GE (1944) indicates the waste streams generated during the final purification steps were
5 transferred back to the 224 Building for disposal through the underground tanks, other records
6 indicate these waste streams may have been routed to nearby cribs for disposal.

7 In summary, the bismuth phosphate process produced a high-activity waste stream containing
8 essentially all of the uranium, and three lower activity waste streams that initially went to
9 underground waste tanks but much of the supernates were later transferred to the soil column.

B3.0 URANIUM RECOVERY PROCESS OPERATIONS

The U Plant became operational in 1944 and included the 221-U Canyon Building, 224-U Concentration Building (later modified and known as the UO$_3$ Plant), and the same ancillary facilities as the T and B Plants. U Plant was based on the design of T and B Plants, and was initially used to train personnel for plutonium recovery using the bismuth phosphate process operations conducted in T and B Plants. During the training phase, only water was used in the plant systems and no waste streams were generated. However, in 1951, U Plant was modified to recover uranium from metal waste that was to be retrieved from the underground storage tanks. As previously noted, the metal waste was the high-uranium waste stream produced during the bismuth phosphate process operations. The uranium recovery process was based on the same solvent extraction chemistry that was later used in the PUREX process. The recovery of uranium from metal waste operated from 1952 to 1958. Recovery of uranium from metal waste involved the following three operational phases (GE 1951a):

- Removal of bismuth phosphate metal waste from underground storage tanks followed by nitric acid dissolution of the solids in ancillary vault tanks and transfer of the high uranium supernate to U Plant

- Recovery of the uranium from dissolved metal waste using a continuous feed solvent extraction process[1]

- Conversion of the uranium nitrate product into uranium trioxide powder for shipment offsite.

Overviews of the uranium recovery process are found in Anderson (1990) and Agnew (1997).

B3.1 METAL WASTE STORAGE AND SLUICING HISTORY [2]

Metal waste was the high fission product waste stream coming from the recovery of plutonium using the bismuth phosphate process that operated in B and T Plants. This stream also included essentially all of the uranium coming from the processed irradiated uranium fuel rods. From 1944 through 1952 at B Plant operations and from 1944 through 1956 at T Plant operations, this high-uranium waste stream had been isolated in specific tanks within the available tank farms (i.e., B, BX, BY, C, T, TX, and U). Groups of tanks were interconnected so that waste would cascade from tank to tank to allow filling of multiple tanks with a single inlet to the first tank in the cascade series. Three-tank cascades series receiving metal waste were found in B, C, T, and U tank farms, while four-tank cascades were found in TX tank farm, and six-tank cascades were created by interconnecting three tanks in the BX tank farm with three tanks in the BY tank farm. By the time the uranium recovery program began, there were approximately 60 underground tanks filled with metal waste. In preparation for the uranium recovery program, there was extensive characterization of the metal waste stored in the underground tanks (GE 1951a).

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[1] This process is generally referred to as the TBP process in documents prepared during the 1950s. Agnew (1997), in defining process waste streams, refers to this process as the uranium recovery (UR) process to minimize confusion with the PUREX waste streams later produced as both processes used the same TBP solvent extraction systems.

[2] The discussion in this section is based on GE (1951a) and Rodenhizer (1987).

1 By filling tanks using the cascade system, the majority of solids settled out into the first tank of
2 the cascade. A much smaller quantity of solids was collected in the second tank; the rest of the
3 tanks in the cascade series contained essentially all supernate. Approximately 75% of the
4 uranium was present in the solid phase according to the *Uranium Recovery Technical Manual*
5 (GE 1951a).

6 The sluicing methodology used for removing the metal waste from the tanks involved pumping
7 the metal waste supernate back into the SSTs containing significant quantities of uranium to
8 break up the solids. This slurry could then be pumped from the tanks. After retrieval from the
9 SSTs, the metal waste was initially transferred to a vault system where the uranium solids were
10 dissolved in nitric acid, and routed to the U Plant via a network of underground pipes and
11 diversion boxes. The sluicing history associated with metal waste recovery from the SSTs is
12 extensively documented (Rodenhizer 1987).

13 Currently, it is unclear as to how much metal waste was left in the tanks at the completion of the
14 sluicing process. Although historical records indicate the tanks were "clean" after completion of
15 the metal waste sluicing campaigns, Agnew (1997) indicates that significant quantities of metal
16 waste or "heels" must have been left in some tanks. If metal waste heels were left in the SSTs, it
17 is likely that heels would be found in the first tank of the cascade series because the first tank is
18 where most of the solids resided, and where the tanks reached the highest temperature (i.e., the
19 tanks containing larger amounts of fission product solids).

20 ## B3.2   URANIUM RECOVERY PROCESS PLANT OPERATIONS

21 The recovery of uranium from the stored metal waste was one of the first applications of a
22 solvent extraction technology in nuclear materials processing. This technology provided a
23 number of advantages over the previous precipitation-based technology. A solvent extraction
24 technology could operate on a "continuous" rather than a "batch" basis, and it was possible to
25 recycle, recover, and reuse many of the chemicals utilized in the process thus minimizing the
26 waste volumes and process costs. For example, the organic solvents used in the extraction
27 process were cleaned or "washed" and reused. Nitric acid was recovered from a number of steps
28 in the process and sent through a fractionalization process and reused. And finally, aqueous
29 waste streams could be pre-concentrated prior to returning them to the underground storage tanks
30 or routing to vadose zone disposal sites if they were within disposal specifications.

31 Initially, it was anticipated that the volume of liquid waste coming from the uranium recovery
32 process operations would be equal to the volume of metal waste removed from the tanks.
33 However, that assumption proved to be incorrect; the volume of waste returned to the tanks was
34 twice the volume of metal waste originally recovered from the tanks.

35 The uranium recovery process used the preferential extractability of uranyl nitrate $[UO_2(NO_3)_2]$
36 by TBP to separate uranium from trace levels of plutonium and fission products associated with
37 the bismuth phosphate metal waste stream. Uranyl nitrate was the product of the acid dissolution
38 of the uranium-bearing solids in the metal waste and is very soluble in an aqueous solution.
39 However, uranyl nitrate forms an organic-soluble complex with TBP that, in the presence of a
40 salting-out agent, can be transferred into an organic phase, leaving the plutonium and fission
41 products in the aqueous phase. Nitric acid was used as the salting agent. The organic phase was

1 a mixture of approximately 20% TBP and the remainder a C_{11} to C_{13} fraction of normal alkane

2 diluent. At the Hanford Site, this n-alkane fraction is commonly labeled as "normal paraffin

3 hydrocarbon," "NPH," "kerosene," or "AMSCO." The organic diluent was needed to create a

4 density of less than 1 for the organic phase.

5 One of the major concerns during the uranium recovery process was the potential co-extraction

6 of trace levels of plutonium with the uranium product. Any plutonium in the +6 or +4 oxidation

7 state would have extracted with the uranium. Thus, care was taken to maintain plutonium in the

8 +3 oxidation state by the addition of ferrous sulfamate during the initial uranium extraction step.

9 Ferrous ions were added to reduce and maintain plutonium to the +3 oxidation state and

10 sulfamate ions were added to react with any nitrite that might be present. This suppressed the

11 nitrite oxidation of plutonium(III). The uranium exited the first extraction column with the

12 organic phase, while the fission products and transuranics remained in the aqueous phase.

13 This aqueous phase was further processed and then returned to the underground storage tanks.

14 The uranium-bearing organic phase was routed to another solvent extraction column where it

15 was contacted with an aqueous stream lacking the "salting" agent. Under these conditions, the

16 uranium transferred back into the aqueous phase as uranyl nitrate. The uranium-bearing aqueous

17 phase was routed to the UO_3 Plant for conversion of the uranium nitrate solution into uranium

18 oxide. The uranium oxide was then shipped offsite.

19 **B3.3 UO_3 PLANT OPERATIONS**

20 The final step of the uranium recovery operations was the conversion of UNH to uranium

21 trioxide (UO_3). After evaporation of the solvent water,[3] the UNH was converted to UO_3 by

22 thermo-decomposition or calcinations. The calcination process liberated water vapor and various

23 oxides of nitrogen. The off-gas was scrubbed with nitric acid to dissolve UO_3 fines and nitric

24 acid was recovered for reuse in the fractionalization operation. The majority of the recovered

25 nitric acid was reused. Aqueous condensates from the calcinations process were routinely routed

26 to soil disposal sites.

27 This process was originally performed in batch operations within the 224-U Building. In 1957,

28 the batch conversion of UNH to UO_3 was renovated. The two original calcinators were replaced

29 with six larger ovens that could operate in a continuous-flow manner. The modified UO_3 Plant

30 operated from 1958 until 1972 converting uranium nitrate received from PUREX and REDOX

31 into UO_3 powder. The UO_3 Plant resumed operations in 1984 to process recovered uranium

32 nitrate from PUREX and ceased operations in 1988.

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[3] Solvent water represents the unbound water in solution rather than the bound water coordinated to the UNH
molecule.

B4.0 WASTE MANAGEMENT ISSUES ASSOCIATED WITH THE BISMUTH PHOSPHATE AND URANIUM RECOVERY PROCESSES

During both the plutonium recovery using the bismuth phosphate process and during the subsequent uranium recovery process operations, extensive quantities of tank wastes and plant process waste streams were discharged to the soil column in and around the 200 Areas. High and modest activity waste streams were originally sent to underground storage tanks, but large quantities of tank waste supernate were later discharged to the soil column (Waite 1991). Lower activity waste streams were routed to cribs, injection wells, and drain fields, while uncontaminated cooling waters were routed to surface impoundments or "swamps." Although this section provides a brief outline of the intentional discharges of higher activity waste to the soil column during the bismuth phosphate and uranium recovery operations, more extensive analyses of waste discharges to the soil column are available in a series of aggregate area management study reports:

- *U Plant Source Aggregate Area Management Study Report* (DOE RL 1992a)
- *T Plant Source Aggregate Area Management Study Report* (DOE-RL 1992b)
- *B Plant Source Aggregate Area Management Study Report* (DOE RL 1993a).

In addition, inventory estimates have been made for many of the discharges to the soil column (Corbin et al. 2005).

Liquid wastes generated by the bismuth phosphate process at T and B Plants that contained high-activity levels were sent to underground storage tanks within the T, TX, TY, B, BX, BY, C, and U tank farms. The waste was routed through a series of collection and transfer tanks, diversion boxes, vaults, and extensive piping systems. With some waste types, initial storage allowed many radionuclides to settle out of solution and form sludge. The remaining liquid supernates were often discharged to the soil column in various engineered facilities such as cribs, drain fields, specific retention trenches, and injection/reverse wells.

Cribs and drain fields were designed to percolate process wastes containing higher activities and larger amounts of constituents into the ground without exposure to the open air. French drains were usually constructed of steel or concrete pipe and were either open or filled with gravel. Cribs were shallow excavations that were either backfilled with permeable material or held open by wooden structures. Cribs usually had an additional layer of an impermeable substance, which allowed the waste to flow directly into the backfilled material, or covered space, and percolate into the vadose zone soils some 200 or 300 ft above the water table. Cribs and drain fields were designed to receive liquid until a specific retention volume or radionuclide capacity was met. There were a number of designs for cribs including:

- Underground caverns constructed by cross stacking 12 in. × 12 in. × 20 ft timbers
- Circular concrete culverts
- Rectangular concrete box structures
- Buried gravel and sand.

Specific retention trenches are shallow, long, narrow, and unlined excavations. Trenches received limited quantities of liquid wastes that were usually higher in activity than the wastes sent to cribs. Often, trenches were located in close proximity to other trenches. After addition of

1   wastes to the specific retention trenches, they were backfilled with the excavated soil.
2   The volume of fluids discharged to specific retention trenches was limited to approximately 10%
3   of the available soil pore volume between the trench bottom and the groundwater table.

4   Reverse injection wells were usually encased holes with the lower end perforated or open to
5   allow liquid to seep to the soil column. These units injected waste into the vadose zone soil at
6   depths greater than the other disposal sites. Injection wells were used for the disposal of "early"
7   liquid wastes from T, B, U, and Z Plants. There was only limited use of reverse injection wells
8   in the 200 Areas in the late 1940s.

9   Waste from the cooling water and steam condensate streams contained very low levels of either
10 radionuclide or chemical waste constituents. Thus, these streams were combined and sent to
11 large surface impoundments. The impoundments were known as swamps or ponds, and waste
12 was routed from the processing facilities to them through piping and open, unlined ditches.

13 **B4.1   BISMUTH PHOSPHATE PROCESS WASTES DISCHARGE**

14 As previously discussed, the uranium or "metal" remained with the majority of activation and
15 fission product waste streams and was routed to underground storage tanks. Approximately
16 7,000 metric tons of uranium was processed at the T and B Plants during the 12 years of bismuth
17 phosphate operations. Beginning in 1945, 224 wastes and cell 5/6 drainage supernates were
18 routed to underground tanks, then the supernates were discharged to nearby cribs and to injection
19 wells in both the 200 East and West Areas. Initially, both the high-level and modest activity
20 waste streams were stored in the underground tank storage system. However, available tank
21 space remained limited even after the construction of four new 12-tank farms in the late 1940s
22 and early 1950s. The chronic shortage of tank space led to the decision to discharge lower
23 activity tank supernates to the soil in and around the 200 Areas. By the mid 1950s, both 1C
24 (1st Cycle) and 2C (2nd Cycle) waste, and scavenged uranium recovery waste were being
25 routinely discharged to cribs and specific retention trenches (Waite 1991).

26 From the beginning of Hanford Site operations, the 224 waste supernates were discharged to the
27 soil column. Beginning in 1948, 2nd Cycle supernates were discharged from underground tanks
28 to nearby cribs in both the 200 East and West Areas. Even with the routine disposal of
29 2nd Cycle and 224 waste supernates to engineered vadose zone facilities, tank waste volumes
30 continued to increase. In 1951, two atmospheric pressure evaporators began operations, the
31 242-B Evaporator in the 200 East Area and the 242-T Evaporator in the 200 West Area.
32 The initial focus for these two evaporators was the reduction of the volume of 1st Cycle wastes
33 stored in the underground tanks. Later, uranium recovery process wastes were also concentrated
34 in both the B and T Evaporators. Anderson (1990) notes that a little over 7 million gal of tank
35 space was recovered through 242-B Evaporator operations in the 1950s and a little over
36 9 million gal of tank space recovered through 242-T Evaporator operations in the same time
37 period. It should be noted that the 16 million gal of recovered tank space represent the capacity
38 of an additional 32 500,000-gal underground storage tanks.

39 Beginning in 1954, bismuth phosphate 1st Cycle wastes were routed from the underground
40 storage tanks to specific retention trenches in both the 200 East and 200 West Areas. In addition,
41 1st Cycle waste concentrated in the 242-B and 242-T Evaporators into what was called

1 "evaporator bottoms" was also discharged to specific retention trenches in the 200 East and
2 West Areas (Waite 1991). The cribs and specific retention trenches receiving 1st Cycle waste
3 supernate are in close proximity to a number of tank farms and these discharges potentially
4 impacted tank farm vadose zone regions.

5 **B4.2 URANIUM RECOVERY PROCESS WASTE DISCHARGED TO PONDS, CRIBS,**
6 **AND SPECIFIC RETENTION TRENCHES**

7 As previously noted, the uranium recovery process was based on the same extraction chemistry
8 that was later used in the PUREX process. Uranium was recovered in the first solvent extraction
9 column leaving a high-activity aqueous waste stream. The uranium product was further purified
10 in additional solvent extraction columns. The high-activity aqueous waste stream was also
11 further processed and then returned to the underground storage tanks. More dilute waste streams
12 were routed to cribs. A second component of the uranium recovery process was the UO_3 Plant,
13 where uranium nitrate was calcined into uranium trioxide for offsite shipment (GE 1951a).

14 The high-activity aqueous waste streams generated in the uranium recovery process from each of
15 the extraction columns were collected in a receiver tank, neutralized with sodium hydroxide, and
16 concentrated in a waste evaporator. Initially, this concentrated waste was pumped back to
17 underground storage tanks. However, beginning in 1954, this waste stream was treated with
18 nickel(II) and ferrocyanide to reduce the cesium-137 activity sufficiently to allow supernates to
19 be discharged to the soil column.

20 With waste streams being produced from three major operating facilities (T Plant, U Plant, and
21 REDOX), and a fourth facility (PUREX) under construction, the tank space recovery efforts due
22 to B and T Evaporator operations and routine discharges of 2nd Cycle and 224 wastes to the soil
23 column were inadequate. However, the direct discharge of uranium recovery waste to the soil
24 column was unacceptable because the high cesium-137 activity in the uranium recovery waste
25 exceeded the 1950s beta/gamma disposal limits for direct discharge of this waste to the soil.
26 After extensive laboratory work, a process was developed that could adequately reduce the level
27 of cesium-137 in the uranium recovery waste so as to allow its disposal to the soil column.
28 The selected process involved the precipitation of cesium-137 with ferrocyanide in a
29 non-stoichiometric (sodium-cesium-nickel-ferrocyanide) solid. The "scavenging process" also
30 removed about 20% of the strontium-90.

31 Beginning in late 1954, high-activity wastes being generated in the U Plant were scavenged with
32 the ferrocyanide. Scavenged uranium recovery waste was transferred to BY farm tanks, solids
33 allowed to settle, and then the supernate transferred to the BY cribs. Approximately 9 million
34 gal of scavenged waste were discharged to seven BY cribs (Waite 1991). After the BY cribs
35 reached their limits, this waste stream was sent to the BC cribs and trenches, where
36 approximately 30 million gal were discharged to 6 cribs and 16 specific retention trenches
37 (Waite 1991).

38 Some uranium recovery wastes produced prior to the implementation of the ferrocyanide
39 scavenging process in U Plant were stored in 200 East Area tanks. An in-tank scavenging
40 process was implemented in the CR vault tanks to remove sufficient cesium-137 so these wastes
41 would meet radionuclide limits, thus allowing them to be discharged to the cribs. The in-tank

1   scavenging process used a number of C farm tanks and scavenged supernates were discharged in
2   BC cribs and trenches.

3   The scavenging process was implemented in T Plant to scavenge bismuth phosphate 1st Cycle
4   waste, which was discharged in specific retention trenches near the T and TX-TY waste
5   management areas.

B5.0   REDOX PROCESS

Construction of the S Plant was completed in September 1951 and the REDOX process was initiated in January 1952. This process introduced a number of innovations in uranium fuel reprocessing that included:

- Recovery of both plutonium and uranium, and some other isotopes of interest from the irradiated uranium fuel

- Continuous, multi-stepped, liquid-liquid solvent extraction based process

- Organic solvent (methyl isobutyl ketone) was cleaned and reused

- Nitric acid was recovered from off-gases, recycled, and reused

- Volume of waste produced, per metric ton of uranium processed, was significantly less than the volume produced in the bismuth phosphate and uranium recovery processes

- Sodium nitrate was recovered from underground storage tanks for reuse in the fuel decladding process.

There were three major issues associated with the REDOX process.

- The transfer of plutonium and uranium complexes into the organic phase required high levels of aluminum nitrate in the aqueous phase. This amount of aluminum led to waste management problems.

- The stability of the organic solvent in nitric acid was a potential hazard.

- The concentrated high-level waste stream resulting from the REDOX process caused the waste in the underground tanks to reach its boiling point. Boiling wastes were difficult to manage due to their heat loads and their impacts on the structural integrities of tanks.

Because of the enhanced efficiency of the REDOX process, the bismuth phosphate process operating in B Plant was put on standby in 1952 and later terminated. However, the bismuth phosphate process in T Plant continued operations until 1956.

Although the REDOX process was based on a radically different chemical separations technology, there were many, very similar chemical oxidation-state manipulation steps in both the REDOX and bismuth phosphate processes. The REDOX process included the following major components:

- Fuel decladding
- Fuel dissolution
- Separation of the uranium and plutonium from fission products
- REDOX process waste management.

Overviews of the REDOX process are found in Anderson (1990) and Agnew (1997). Detailed information on the REDOX process can be found in the *REDOX Technical Manual* (GE 1951b). A detailed analysis of the liquid waste streams from REDOX that were intentionally discharged to the soil column can be found in the *S Plant Source Aggregate Area Management Study Report* (DOE-RL 1993b).

B5.1   URANIUM FUEL DECLADDING

During REDOX process operations, the aluminum-clad fuel was decladded using the same chemical reactions that were utilized in the bismuth phosphate process. The only difference was the way the "aluminum cladding" waste stream was handled. In the REDOX process (and later in the PUREX process), the aluminum cladding waste stream was segregated in underground storage tanks. Although this was a relatively low-activity waste stream, it could not be discharged to the soil column because of the reactivity of aluminum ions with the soil. As in the bismuth phosphate process, less than 500 gal of aluminum cladding waste was generated per metric ton of uranium fuel reprocessed. However, the aluminum cladding waste stream was frequently used to replace water that had evaporated from REDOX high-level waste tanks.

In the early 1960s, a different fuel cladding material was introduced. In December 1962, the first tests of the zirconium-clad Plutonium Recycle Test Reactor (PRTR) fuel dissolution were completed at S Plant. The PRTR fuels were plutonium-aluminum and plutonium-uranium alloy fuel rods that had been encased with a zirconium-alloy coating (Zircaloy) and irradiated in the PRTR located in the 300 Area. PRTR fuels were reprocessed at the S Plant from 1963 through 1966, along with the normal aluminum-clad fuels. Beginning in 1965, zirconium-clad N-Reactor enriched uranium fuels were processed at S Plant. The dissolution of zirconium-cladding required a much more reactive chemical process than had been required for aluminum cladding dissolution. A boiling solution of ammonium hydrogen fluoride/ammonium nitrate was used to dissolve the zirconium-cladding. Because of the reactivity of the cladding dissolution process, congruent dissolution of the irradiated fuel was more problematic than it had been with the aluminum decladding process. Thus, it is likely the zirconium-cladding waste going to the underground storage tanks contained more transuranic elements than the aluminum cladding waste.

B5.2   FUEL DISSOLUTION

As with the bismuth phosphate process, after decladding, the uranium fuel was dissolved in concentrated nitric acid, creating an acidic solution containing primarily uranyl nitrate, plutonium(IV) and fission products. This dissolver solution was treated with potassium permanganate and sodium dichromate to oxidize all of the plutonium to the +6 oxidation state, and to oxidize the ruthenium-106 to the volatile ruthenium tetraoxide. The ruthenium-106 was then stripped into the off-gas system. The manganese dioxide produced from the permanganate oxidation reactions functioned as a carrier for removing niobium and zirconium. At this point, after the removal of solids by centrifugation, both the uranium and plutonium were in the +6 oxidation state. The dissolver solution was then ready for separations and recovery of plutonium and uranium.

B5.3   SEPARATION OF THE URANIUM AND PLUTONIUM FROM FISSION PRODUCTS

In the +6 oxidation state, both plutonium and uranium will transfer from a slightly acidic, high ionic aqueous solution into the organic phase (methyl isobutyl ketone). This separations chemistry became the basis for separating plutonium and uranium from high-activity fission products using a series of counter-current liquid-liquid extraction columns. The dissolver solution, containing both metals in the +6 oxidation state, was fed into a counter-current

1   liquid-liquid extraction column at the mid-point. Uranium and plutonium were extracted into the
2   organic phase, while almost all of the fission products remained in the aqueous phase.
3   After additional processing (discussed below), the aqueous high-level waste stream went out to
4   the underground storage tanks. The organic phase then moved to a second column where
5   plutonium was separated from the uranium.

6   The high ionic strength aqueous phase in the second liquid-liquid extraction column contained
7   ferrous and sulfamate ions that, in combination, reduced the plutonium to a +3 oxidation state, at
8   which point the plutonium(III) transferred back into the aqueous phase. The uranium, still in the
9   +6 oxidation state, remained in the organic phase and moved to a third liquid-liquid extraction
10   column. In this third column, the organic phase was contacted with a dilute nitric acid solution.
11   In the absence of a salting-out agent, the uranium transfers back into the aqueous phase. At this
12   point, the organic solvent was cleaned and then reused in the process. The crude plutonium and
13   uranium products were then separately purified through additional passes through solvent
14   extraction columns. The purified uranium nitrate was transferred from S Plant to the UO_3 Plant
15   for conversion to uranium oxide. Purified plutonium nitrate was transferred to the 231-Z Facility
16   for additional processing.

17   **B5.4   REDOX PROCESS WASTE MANAGEMENT ISSUES**

18   The REDOX process produced a number of waste streams including a high-level waste stream
19   that went to the underground storage tanks, an aluminum cladding waste stream that also went to
20   the tanks, dilute radioactive liquid waste streams that went to the soil column, nonradioactive
21   liquid waste streams discharged to surface ponds, and major off-gas streams.

22   **B5.4.1   REDOX High-Level Waste Streams**

23   Fission product waste streams coming from the solvent extraction columns were combined and
24   the volume reduced by a factor of 2 before being neutralized and routed to the underground
25   storage tanks. The goal was to minimize the volume of high-level waste being transferred to the
26   underground storage tanks. This volume reduction process was limited by aluminum ion
27   solubility. It was known that once the aluminum salts began to precipitate in the waste stream,
28   the resulting fluid became unpumpable. Since the solubility of aluminum salts is a complex
29   function of both hydrogen ion concentration and temperature, detailed operational procedures
30   were required to handle this waste stream (GE 1951b).

31   The REDOX high-level waste stream was considerably more concentrated than the similar waste
32   stream coming from the bismuth phosphate process. The highly concentrated high-level waste
33   streams led to the need to control heat loads in the tanks storing these wastes. Although
34   high-heat loads in the REDOX high-level waste tanks were seen as a mechanism for further
35   volume reduction of these wastes, there were significant operational problems in managing these
36   wastes. The high-heat loads, particularly in the SX tanks, undoubtedly led to premature failure
37   of a number of the underground storage tanks (Field and Jones 2005). Condensates from the
38   high-heat tanks were either recycled to maintain desired tank volumes or discharged to the soil
39   column.

B5.4.2   Dilute Liquid Radioactive Waste Streams

A number of waste streams generated during REDOX operations were diluted enough to be discharged to the soil column. These waste streams included condensates from the high-level waste volume reductions as well as spent organic solvents. Large volumes of nonradioactive liquid waste streams were discharged to surface impoundments (initially called swamps) in or near the 200 West Area.

B5.4.3   Major Off-Gas Waste Streams

There were three major off-gas waste streams emanating from the S Plant. The first was associated with the cladding dissolution process. The chemical contaminates in this waste stream were primarily nitrogen oxides and ammonia. The second off-gas waste stream originated from the uranium fuel dissolution process and included oxides of nitrogen, volatile radionuclides such as iodine-129, iodine-131, and trace amounts of radioactive argon and krypton. This off-gas stream was routed through a "silver reactor" column that was designed to retain or delay atmospheric release of the iodine-131.

The third major off-gas waste stream was associated with the removal of ruthenium-106 for the dissolver solution. Significant reduction of ruthenium-106 in the dissolver solution prior to the recovery of plutonium and uranium was required to minimize ruthenium-106 contamination in both product streams. As previously noted, the solution generated from the dissolution of the irradiated fuel was treated with strong oxidants to convert the ruthenium to the volatile ruthenium tetraoxide species. The ruthenium tetraoxide was captured in a sodium hydroxide scrubber system and routed to the underground storage tanks.

Liquid waste generated by the off-gas treatment systems, including 291-S stack drainage, various condensed process drainages, and liquid effluents from the silver reactor, condensers, and filters were collected and routed to the condensate stripper. There, the organics (primarily hexone and hexone impurities) were stripped (by steam) from the aqueous waste. The organic vapors were routed to an organic distillation column for further treatment. The resulting aqueous waste stream was sent to a condensate evaporator and then to the 216-S cribs.

More detailed information on intentional discharges of REDOX low-level waste streams to the soil column can be found in DOE-RL (1993b), and current inventory estimates can be found in Corbin et al. (2005).

B6.0   PLUTONIUM-URANIUM EXTRACTION (PUREX) PROCESS (1956 TO 1972 AND 1983 TO 1989)

The PUREX process was the third generation fuel reprocessing technology. Approximately 75% of the uranium fuel reprocessed at the Hanford Site was processed at PUREX. The PUREX process came online in 1956 in the A Plant, and provided a number of advantages over the REDOX process:

- Separation process was more efficient.

- Organic solvent and extractant used in the PUREX process were more stable to nitric acid decomposition than that in use in the REDOX process.

- Nitric acid, rather than aluminum ions, could be used as a salting-out agent.

- Waste volumes per metric ton of uranium processed decreased by a factor of 3 or more, when compared to the REDOX process.

A number of irradiated fuel-types were reprocessed using the PUREX process in A Plant. These included aluminum-clad natural and enriched uranium fuel, enriched uranium fuel, and aluminum-clad throrium-232 targets. As with the REDOX process, uranium, plutonium, and neptunium were recovered in the PUREX process, as well as smaller quantities of other isotopes. The PUREX process included a nitric acid recovery system and a system for cleanup and reuse of the organic solvents. As with both the bismuth phosphate process and REDOX, the fundamental separations mechanisms in the PUREX process depended on maintaining selected oxidation states for the chemicals of interest so as to selectively control their movement of specific species between the organic and aqueous phases. The PUREX process used a nitric acid-based aqueous phase and an organic phase composed of TBP in an n-alkane diluent. The use of nitric acid as the salting agent allowed considerably more flexibility in separations systems and waste management operations.

With the exception of the fuel decladding and dissolution processes, which operated in batch operations, the PUREX process was a continuous process. The process steps include:

- Fuel decladding and dissolution
- Separation and recovery of uranium and plutonium from fission products
- Further purification cycles of the uranium, plutonium, and other isotopes of interest
- Solvent recovery, treatment, and recycle
- Nitric acid recovery, fractionalization, and recycle
- Back-cycle waste treatment system and process condensate recycle.

Three major waste streams from PUREX were routed to the underground storage tanks: 1) a highly concentrated high-level waste, 2) cladding waste from both the aluminum- and zirconium-clad fuels, and 3) a waste stream generated as part of the organic solvent cleanup step. As with previous processes, a number of lower activity waste streams from PUREX were routed to cribs. Finally, large volumes of uncontaminated cooling water were routed to surface impoundments.

Overviews of the PUREX process are found in Anderson (1990) and Agnew (1997). A detailed analysis of the liquid waste streams from PUREX that were intentionally discharged to the soil

1 column can be found in the *PUREX Source Aggregate Area Management Study Report*
2 (DOE RL 1993c).

3 **B6.1   FUEL DECLADDING AND DISSOLUTION**

4 During PUREX process operations, the aluminum-clad fuel was declad using the same chemical
5 reactions that were used in the bismuth phosphate and REDOX processes. As with the REDOX
6 process, the aluminum-cladding waste stream generated at A Plant was segregated into
7 underground storage tanks. Although this was a relatively low-activity waste (LAW) stream, it
8 could not be discharged to the soil column because of the reactivity of aluminum ion with the
9 soil. Less than 500 gal of aluminum-cladding waste was generated per metric ton of uranium
10 fuel reprocessed at PUREX. The aluminum-cladding waste stream was frequently used to
11 replace water that had evaporated from PUREX high-level waste tanks.

12 In December 1962, the first tests of the dissolution of zirconium-clad fuel were completed at
13 S Plant. Zirconium-clad fuels were reprocessed in S Plant from 1963 through 1966.
14 Beginning in 1967, zirconium-clad N-Reactor enriched uranium fuels were processed at A Plant.
15 The dissolution of zirconium-cladding required a much more reactive chemical process than had
16 been required for aluminum-cladding dissolution. A boiling solution of ammonium hydrogen
17 fluoride/ammonium nitrate was used to dissolve the zirconium-cladding. Because of the
18 reactivity of the cladding dissolution process, congruent dissolution of the irradiated fuel was
19 more problematic than it had been with the aluminum decladding process. Thus, it is likely the
20 zirconium-cladding waste going to the underground storage tanks contained more transuranic
21 elements than the aluminum-cladding waste.

22 As with bismuth phosphate process and REDOX, after decladding, the uranium fuel was
23 dissolved in concentrated nitric acid, creating a metal solution containing primarily uranyl
24 nitrate, plutonium(IV), and fission products.

25 **B6.2   SEPARATION OF URANIUM AND PLUTONIUM FROM FISSION PRODUCTS**

26 As with REDOX, the PUREX process was based on counter-current solvent extraction
27 separations, where the plutonium(IV) and uranium(VI) were initially separated from fission
28 products followed by separation of plutonium from uranium. However, the PUREX solvent is
29 quite similar to the solvent system used in the uranium recovery process. In the organic phase in
30 the PUREX process, TBP was used as the extractant and a $C_{11} \sim C_{13}$ n-alkane was used as the
31 dilutant. In Hanford Site documentation, the $C_{11} \sim C_{13}$ n-alkane is generally referred to as
32 "normal paraffin hydrocarbon," "NPH," "kerosene," or by a trade name. TBP offered several
33 advantages over the organic solvent/extractant, methyl isobutyl ketone, used in the REDOX
34 process. First, TBP was much more stable in nitric acid than methyl isobutyl ketone.
35 This allowed nitric acid to be used as the "salting-out agent" rather than aluminum salts.
36 The ability to use nitric acid as the salting-out agent provided considerably more flexibility in
37 operating the solvent extraction columns, recovering and reusing chemicals, and in handling
38 aqueous waste streams.

39 The second major advantage with using TBP as the extractant was that plutonium was
40 transferred into the organic phase while in the +4 oxidation state, and uranium was extracted

1   while in the +6 oxidation state. These were the predominant oxidation states for these two
2   elements in the dissolver solution; thus, there was no need for the use of strong oxidizing agents
3   such as dichromate or permanganate to oxidize the plutonium to the +6 oxidation state.

4   Sodium nitrite was added to the dissolver solution prior to entering the first extraction column to
5   stabilize the plutonium in the +4 oxidation state. Both plutonium(IV) and uranium(VI) were
6   transferred into the organic phase, while almost all of the fission products remained in the
7   aqueous phase. Plutonium was separated from uranium by reducing the plutonium to a
8   +3 oxidation state by adding ferrous sulfamate to the aqueous phase. In this column,
9   plutonium(III) was transferred back to the aqueous phase, while uranium(VI) remained in the
10 organic phase. Uranium was separated in another column by contacting the organic phase with
11 a dilute nitric acid aqueous phase.

12 The impure plutonium and uranium products were then separately purified through additional
13 passes through solvent extraction columns. The purified uranium nitrate was transferred from
14 the A Plant to the UO_3 Plant for conversion into uranium trioxide. Purified plutonium nitrate
15 was transferred to the 231-Z Facility for additional processing.

B6.2.1   Solvent Treatment and Recycle and Nitric Acid Recovery

17 As with the REDOX process, organic solvents were cleaned and reused until such time that
18 separations efficiency decreased below operating specifications, at which time the solvents were
19 discarded. The major waste streams generated during solvent cleanup were the organic-wash
20 waste and the spent solvent waste stream. The organic-wash waste went to the underground
21 storage tanks and the spent solvents appear to have been discharged to cribs.

B6.2.2   PUREX Process Waste Management Issues

23 The major issues associated with the PUREX process were associated with handling the highly
24 radioactive high-level waste stream. With PUREX, the separations efficiencies improved to the
25 point where the high-level waste stream volumes were around 400 gal per metric ton or less.
26 Thus, managing the heat loads in the high-level waste tanks became paramount. From 1956
27 through 1968, PUREX high-level wastes were routed to specially designed tanks in the A and
28 AX tank farms where short-lived radionuclides were allowed to decay. Because of decay heat,
29 the wastes in these tanks reached their boiling point and wastes were allowed to self-concentrate.
30 However, waste compositions were closely monitored and tank waste condensate or other low
31 active waste types were added back to the tanks to maintain desired liquid levels. Loss of
32 PUREX high-level wastes to the soil column was minimal, and included small leaks in some
33 A and AX farm tanks, as well as piping systems leaks. By far the largest impacts to the
34 environment from PUREX operations resulted from intentional discharges of LAW to cribs and
35 high-volume discharges of uncontaminated water to surface impoundments.

36 More detailed information on intentional discharges of PUREX low-level waste streams to the
37 soil column can be found in DOE-RL (1993b), and current inventory estimates can be found in
38 Corbin et al. (2005).

B7.0   HANFORD SITE ISOTOPE RECOVERY PROGRAMS

B7.1   INTRODUCTION

Although the major focus of the Hanford Site operations was the production of plutonium, there were specific programs that centered on the recovery of radioisotopes other than those of plutonium and uranium. Early small-scale isotope recovery activities included isotopes such as lanthanum-140, tritium, cobalt-60, and small quantities of strontium-90, cesium-137, and technetium-99. Large scale isotope recovery activities focused on the recovery of uranium-233 from thorium-232 targets in PUREX, and the removal of cesium-137 and strontium-90 from selected tank waste streams in B Plant so as to allow their conversion to solids.

Very early in the site operations, the Hanford Site reactors were used to produce small quantities of radioisotopes such as lanthanum-140, tritium, and cobalt-60. However, these activities do not seem to have contributed significant quantities of wastes to the underground storage tanks in the 200 Areas. In the early 1960s, there was considerable interest in the recovery of strontium-90 for use as heat sources. To a lesser extent, there was interest in recovering other radioisotopes such as cesium-137 and technetium-99. Recovery methods for recovery of these isotopes were developed in 300 Area research and development laboratories, and transferred to the Hot Semi-Works Facility in the 200 East Area. Other components of these separation processes were conducted in S Plant, A Plant, and B Plant. High-level waste streams from these isotope recovery programs went primarily to C farm tanks and low-activity streams went to cribs. Neptunium-237 was recovered from both REDOX and PUREX high-level waste streams. The wastes from this process were incorporated with other high-level waste streams coming from those facilities.

The major radioisotope recovery programs at the Hanford Site were driven by tank waste management requirements. By the late 1950s, it was clear that a number of underground storage tanks had likely leaked and the long-term storage of large volumes of liquid radioactive wastes in these tanks was untenable. Hanford Site contractors were directed to convert liquid radioactive waste into solids as soon as practicable. After extensive engineering studies, the decision was made to convert liquid wastes into saltcake by evaporating the liquid phase. The problems associated with converting liquid wastes to saltcake have long been appreciated. These included aluminum solubility issues (Barney 1976) and controlling heat production in stored saltcake (Smith and Tomlinson 1967).

The conversion of high-fission product radioactive waste supernates into saltcake required both a 3- to 5-year cooling-off period to allow short-lived radionuclides to decay (thus, the need for boiling waste tanks in the S, SX, A, and AX tank farms), and removal of a significant amount of the longer-lived, heat-generating radionuclides such as strontium-90 and cesium-137. In 1967, B Plant was reactivated to support an isotope recovery program. Beginning in 1967, PUREX current acid wastes were processed through B Plant for cesium-137 and strontium 90 recovery, prior to the 3- to 5-year cooling-off period. Aged PUREX supernates and sludges were recovered from the tanks and processed through the B Plant for strontium-90 and cesium-137 recovery. The aged REDOX supernates were transferred to the 200 East Area tanks and processed through B Plant for cesium-137 recovery. After cesium-137 removal, much of the REDOX supernates were transferred back to the 200 West Area for saltcake production in the T and S Evaporators. Available tank space in the T, TX, and TY tank farms were filled using the

1   T Evaporator and S, SX, and U tank farms tanks were filled using the S Evaporator.

2   The REDOX sludges, which contained essentially all of the strontium 90, were left in SX tanks

3   and air cooled.

4   After the cesium-137 was removed (or at least greatly reduced) in the aged PUREX supernates,

5   the wastes were transferred to various B, BX, and BY tanks, leading to conversion of supernates

6   into saltcake using the BY farm in-tank solidification (ITS) process. In the ITS process,

7   heater units were installed in three tanks in the BY tank farm. Waste supernates were rotated

8   through the ITS process tanks and out to the B, BX, and BY tanks so as to produce saltcake.

9   The available tank space in the B, BX, and BY tank farms was filled with saltcake using the ITS

10 process. Available tank space in C tank farm was not filled in the ITS process because the

11 C tank farm was located too far away for concentrated waste transfers.

B7.2   ISOTOPE RECOVERY CHEMISTRY

13 The isotope recoveries were accomplished with plant-scale solvent extraction, ion exchange, and

14 precipitation processes (Buckingham 1967). Although some earlier pilot plant isotope recovery

15 operations were completed in PUREX, the Hot Semi-Works Facility, and B Plant, the major

16 isotope recovery program began in late 1967 (Larson 1967, 1968; Buckingham 1967). Initially,

17 the chemical separations processes followed the flowsheets reported by Larson (1967, 1968).

18 However, in practice, the separations efficiencies were not as good as expected (Gasper 1989).

19 Thus, there was a continuing series of process change improvements that appear to have been

20 ongoing throughout the life of this program. It appears that effluent waste streams going back to

21 the tanks had higher activity levels than are suggested by process flowsheet data. Waste transfer

22 records indicate that some B Plant isotope recovery waste streams were routinely commingled

23 with lower activity wastes such as the aluminum-cladding waste or organic-wash waste prior to

24 being converted into saltcake.

25 Cesium-137 was recovered in B Plant from PUREX current acid waste, aged PUREX high-level

26 waste supernate, and REDOX high-level waste supernate. The PUREX current acid waste was

27 transferred directly from A Plant to B Plant; it contained the full compliment of short-lived

28 radionuclides. Rather than directly neutralizing this high-acid waste stream with hydroxide, this

29 waste stream was treated with sugar to consume much of the nitric acid. This process was

30 known as "sugar denitrification" and had the effect of producing a much lower ionic strength

31 feed stream to the ion exchange columns.

32 Initially, an inorganic-based ion exchange resin was used for cesium recovery. Later, an

33 organic-based resin was used. According to Gasper (1989), neither of these ion exchange resins

34 worked as well as flowsheets had indicated. Although poorly documented in the flowsheets, it

35 appears that complexants such as hydroxyl-ethyl-ethylenediamine-triacetic acid (HEDTA),

36 ethylenediamine-tetra-acetic acid (EDTA), and citrate were added to sequester divalent and

37 trivalent cations to enhance cesium recovery (Gasper 1989). The addition of complexants to the

38 cesium recovery feed stream led to mobility of some radionuclides that would otherwise have

39 sorbed strongly to the soil. The increased mobility of certain radionuclides in the cesium

40 recovery waste stream is seen in the 1973 leak from tank T-106.

1   Strontium-90 recovery process in B Plant was much more complex that the cesium-137 recovery
2   process. The strontium recovery system used a liquid-liquid solvent extraction system.
3   The extractant was a mixture of di-2-ehtyhexyl-phosphoric acid and TBP in an n-alkane diluent.
4   Strontium-90 was recovered from two waste streams, the PUREX current acid waste and the
5   aged PUREX sludge recovered from tanks.

6   A number of organic complexants were required for successful recovery of strontium from this
7   feed stream. The resulting strontium-90 waste stream going back to the underground storage
8   tanks contained significant levels of the complexants. The presence of high levels of organic
9   complexing agents led to significant waste management issues in tanks such as tank SY-101,
10   where the decomposition of organic compounds allow the buildup of dangerous levels of
11   hydrogen gas and other reactive gases.

B8.0   HANFORD SITE MATERIALS FINISHING PLANTS

B8.1   INTRODUCTION

A number of processes at the Hanford Site focused on preparing products for shipment offsite or for long-term storage. The major product "finishing" operations was the conversion of the plutonium product from the separations plants into a product suitable for shipment offsite. Plutonium finishing activities at the Hanford Site have a complex history. Uranium recovered in the separations plants as nitrate salt was converted into the oxide for offsite shipment. Large quantities of cesium 137 and strontium-90 recovered from tank wastes were encapsulated for long-term storage. The major finishing operations are discussed below.

B8.2   UO$_3$ PLANT OPERATIONS

The final step of the uranium recovery operations was the conversion of uranium nitrate hexahydrate (UNH) to UO$_3$. This operation was accomplished by calcining the UNH in a batch process within the 224-U Building in what became known as the UO$_3$ Plant. In 1957, the batch conversion of UNH to UO$_3$ was renovated. The two original calcinators were replaced with six newer ones that could operate in a continuous-flow mode. The UO$_3$ Plant operated from 1958 until 1972 converting UNH received from PUREX and REDOX into UO$_3$ powder. Uranium oxide was packaged at the UO$_3$ Plant and shipped offsite. The UO$_3$ Plant resumed operations in 1984 to process UNH from PUREX and ceased operations in 1988.

During the conversion of UNH into uranium oxide, off-gas scrubbers allowed the recovery of nitric acid for reuse in the metal waste dissolution and Uranium Recovery Plant (U Plant) operations. Aqueous condensates from the calcinations process were routed to cribs.

B8.3   Z PLANT PROCESS DESCRIPTION AND HISTORY

The Z Plant complex was used to finish the plutonium product in a variety of processes and operations including isolation and purification of plutonium solutions, production of finished metallic plutonium and plutonium oxides, milling and machining of plutonium oxides, and recovery of plutonium and americium from plutonium scrap materials. Components of the Z Plant operated from 1945 until 2004.

Throughout its lifetime, the Z Plant complex received various types of processed (i.e., uranium and fission products removed) plutonium solutions from each of the 200 Area separations facilities. Most recently, the Z Plant complex was used to stabilize plutonium containing solids, solutions, and incinerator ashes.

Several facilities are associated with the Z Plant complex. These include:

- 231-Z isolation building
- 234-5Z plutonium finishing building
- 236-Z plutonium reclamation facility
- 242-Z americium recovery building
- 241-Z waste treatment and collection building
- 232-Z waste incinerator facility
- 291-Z building and exhaust stack
- 2736-Z and 2736-ZB plutonium storage vaults.

1   In-depth discussion of each facility and their operations can be found in Chapter 2 of
2   *Plutonium/Organic-Rich Process Condensate/Process Waste Group Operable Unit RI/FS Work*
3   *Plan: Includes the 200-PW-1, 200-PW-3, and 200-PW-6 Operable Units* (DOE-RL 2001).

4   At the Z Plant complex, the recovered, purified plutonium was refined to one of several forms
5   depending upon the era and available process. The operations conducted at the Z Plant complex
6   included the following:

7   • Plutonium Isolation: The Plutonium Isolation Facility (PIF) or 231-Z Facility was
8   constructed in 1944. Operations began in 1945 and continued until 1949. At the PIF,
9   plutonium product from both T and B Plants were further decontaminated. Once the
10   plutonium was decontaminated of fission products (americium and lanthanum), it was
11   converted to a nitrate paste and thickened before shipment offsite.

12   • Rubber Glove (RG) line: Operation was transferred to the newly constructed
13   234-5 Building in 1949 and operated until 1953, when it was replaced with remote
14   mechanical operations. The process was an inorganic chemical precipitation to further
15   purify the plutonium and convert it from a nitrate paste to an oxide. Waste generated by
16   this process included hydrofluoric, sulfuric, and nitric acids, as well as peroxide,
17   plutonium, and other transuranic metals.

18   • Remote Mechanical A (RMA) line: The RMA line was constructed within the
19   234-5 Building in 1949 and began operations in 1953. The RMA line operated until the
20   mid 1960s. The process was the same as the RG line chemically; however, the plutonium
21   was handled by remote mechanical means. Thus, the RMA produced the same waste as
22   the RG line.

23   • Remote Mechanical C (RMC) line: The RMC line was constructed in 1957 and began
24   operations in 1960. The RMC operated until 1973 and again from 1985 to 1989.
25   The process was the same as the RG and RMA lines chemically; however, the plutonium
26   was handled remotely by mechanical means, with additional mechanical upgrades to
27   increase operator safety. Thus, the RMC produced the same waste as the RG and RMA
28   lines.

29   • Plutonium Metal Fabrication: Weapons-grade plutonium metal was cut and milled into
30   weapons shapes for assembly into nuclear weapons from the early 1950s to late 1970s.
31   Waste generated by this process included "cutting oil" or fabrication oil, which is a
32   mixture of lard and carbon tetrachloride, as well as other volatile organics used as cutting
33   fluids.

34   • RECUPLEX: This plutonium recovery process operated in the 234-5Z Building from
35   1955 to 1962, at which time the process was terminated after a criticality event
36   (uncontrolled nuclear reaction) occurred within the Plutonium Fishing Plant. The process
37   was based on a solvent extraction similar to PUREX. However, carbon tetrachloride was
38   utilized as the diluent rather than kerosene. Waste generated by this process included
39   hydroiodic, hydrofluoric, sulfuric, and nitric acids, as well as silver, carbon tetrachloride,
40   TBP, plutonium, and other transuranic metals.

- Plutonium Reclamation Facility (PRF): The PRF replaced the RECUPLEX plutonium recovery process in 1964 and was located in the 236-Z Building. The PRF operated from 1964 to 1979 and from 1984 to 1987. It was the same chemically as the RECUPLEX process, just on a larger production scale. Waste generated by this process included hydrofluoric, phosphoric, and nitric acids, along with silver, hydroxyl amines, dibutylbutyl phosphonate (DBBP), carbon tetrachloride, TBP, uranium, plutonium, and other transuranic metals.

- Americium Recovery: An americium recovery process operated in the 242-Z Building between 1964 and 1976. It was shut down in 1976 after an explosion occurred in one of the ion exchange columns. The americium recovery process was based on an initial recovery of americium from a plutonium recovery waste stream using solvent extraction technology followed by further purification using ion exchange columns (Szulinski 1964). The solvent extraction process used DBBP as the extractant and carbon tetrachloride as the diluent. Waste generated by this process included hydrochloric, hydrofluoric, phosphoric, and nitric acids, as well as DBBP, carbon tetrachloride, plutonium, and other transuranic metals.

- Laboratory Operations: The Z Plant analytical and development laboratories were housed in the 234-5Z and 231-Z Buildings. The laboratory provided analytical services and supported research and development activities for the Plutonium Finishing Plant, including quality assurance and quality control for the plutonium processing lines, liquid scintillation counting, and preparation work for solvent extraction tests. Laboratory process wastes were characterized as slightly acidic, low salt radioactive wastes. The pH of these wastes was adjusted to between 8 and 10 in the 241-Z treatment tank prior to disposal.

Processes at the Z Plant complex generated the primary waste streams that fed various soil disposal sites including cribs, trenches, French drains, and injection wells until 1973. After 1973, waste was routed to burial grounds for storage and disposal, and routed to underground storage tanks for processing through the evaporator waste reduction operations. Process condensates from T Evaporator operations were then sent to nearby soil disposal sites such as the 216-T-19 crib.

Z Plant discharges to the soil column have led to significant environmental impacts (*Z Plant Source Aggregate Area Management Study Report* [DOE-RL 1992c]; *200 West Groundwater Aggregate Area Management Study Report* [DOE-RL 1993d]). Some of the slightly acidic waste streams emanating from Z Plant that were routinely discharged to the soil column carried transuranic contaminants that tended to sorb on the soil near the point of discharge. Eventually, this led to concerns about potential criticality accidents associated with certain disposal sites such as the Z-9 crib. These safety concerns led to significant environmental restoration activities at Z-9 crib. In addition, large volumes of spent solvents from the RECUPLEX and PRF operations (primarily carbon tetrachloride) were discharged to cribs near the Z Plant. The large volume of carbon tetrachloride discharges have led to a large groundwater and vadose zone plume in the 200 West Area. Although significant volumes of other organic compounds such as TBP were discharged with carbon tetrachloride, they have not been found in field characterization studies.

1   Cooling water generated from these various processes and operations was routinely discharged to
2   evaporation basins and routed to ponds, ditches, and French drains. In the mid 1990s, this stream
3   was diverted to a manhole collection system and sent to the Liquid Effluent Retention Facility
4   and Effluent Treatment Facility in the 200 East Area.

5   **B8.4   WASTE ENCAPSULATION AND STORAGE FACILITY**

6   Large quantities of cesium-137 and strontium-90 were recovered from process waste streams
7   between 1967 and 1978 in the B Plant Isotope Recovery Program. The strontium-90 and
8   cesium-137 were transferred to the Waste Encapsulation and Storage Facility for final packaging
9   and long-term storage. Strontium-90 was transferred as strontium nitrate, converted to strontium
10 fluoride, sintered, and double-encapsulated in metal containers. Cesium-137 was transferred as
11 cesium carbonate, converted to the chloride salt, and double-encapsulated in metal containers.
12 The encapsulated strontium 90 and cesium-137 are stored in water-cooled basins at the Waste
13 Encapsulation and Storage Facility (Jeppson 1973).

B9.0   HANFORD SITE WASTE VOLUME REDUCTION OPERATIONS

B9.1   INTRODUCTION

The management of the large volumes of liquid wastes coming from the Hanford Site chemical processing facilities has always been a daunting task. Initially, of particular concern were the radioactivity waste streams that required storage in the underground tanks. For this, waste volume reduction was the goal. Later, this goal changed to conversion of liquid wastes into solids that could be safely stored long-term in the SSTs. The latter efforts have led to the state of the current wastes stored in the SSTs.

As previously discussed, one of the major issues during fuel reprocessing at the Hanford Site was the management of the large volumes of liquids generated in the chemical processing facilities. Cooling water and uncontaminated steam condensates were discharged to surface impoundments, low-activity liquid wastes were discharged to subsurface soil disposal sites, and higher activity wastes were routed to the underground storage tanks. However, given that tank space was always limited, management of available tank space was a critical task. A brief overview of evaporator operations is given in Anderson (1990) and Agnew (1997).

B9.2   ATMOSPHERIC PRESSURE EVAPORATORS

In 1949, the concept of concentrating 1st Cycle decontamination wastes from B and T Plants by evaporation was proposed. After favorable engineering and economical feasibility studies, construction of the B and T Evaporators began. Two atmospheric pressure evaporators began operations in 1951, the 242-B Evaporator in the 200 East Area and the 242-T Evaporator in 200 West Area. The initial focus for these two evaporators was the reduction of the volume of 1st Cycle wastes stored in the underground tanks. First cycle supernates were transferred from B and C tank farm tanks to the 242-B Evaporator, and 1st Cycle waste from T, TX, and U tank farms was transferred to the 242-T Evaporator. Later, uranium recovery process wastes and 2nd Cycle wastes were concentrated in both the B and T Evaporators. The B Evaporator was shut down in 1954 and the T Evaporator was shut down in 1956.

During operations, tank supernates were transferred to the evaporator facility, and heated to their boiling point. Volumes were reduced to the point that solids would form and settle upon cooling. After volume reduction in the evaporator, the concentrated supernates were transferred back to the underground tank to cool and form solids (i.e., saltcake). The slightly contaminated evaporator condensates were transferred to nearby cribs for disposal. After the concentrated supernates had cooled and the saltcake had settled, the pumpable liquids were routed back to evaporator for additional liquid reduction.

B9.3   BOILING WASTE TANK OPERATIONS

A different strategy was used in dealing with the reduction of volumes in the REDOX and PUREX high-level waste streams. These waste streams contained sufficient concentrations of short-lived radionuclides such that the decay heal caused these wastes to boil. These waste streams were allowed to self-concentrate in "boiling waste" tanks. Steam condensates from the boiling waste tanks were condensed and some fluid added back to the tanks, as needed, to maintain the desired waste volume, otherwise the condensate was discharged to nearby cribs.

1 In some cases, other lower activity wastes, particularly aluminum-cladding waste and PUREX
2 organic-wash waste, were used to replenish liquids in the boiling waste tanks. Once the
3 short-lived radionuclides had decayed sufficiently such that the waste stopped boiling, the
4 "aged" high-level wastes were transferred to "non-boiling" waste tanks for long-term storage
5 until these wastes could reprocessed and converted to saltcake.

B9.4   IN-TANK WASTE SOLIDIFICATION

7 In the mid 1960s, the decision was made to convert liquids stored in the Hanford Site
8 underground storage tanks into saltcake for long-term storage. This process was begun using
9 another strategy. In 1965, a prototype heater was installed in tank BY-101. In 1966, a second
10 heater was installed in tank BY-102 and a third heater was installed in tank BY-112. During
11 operation of these in-tank heater systems, supernates were transferred into the "heater tank,"
12 volumes reduced, and then transferred to another tank to allow precipitation of saltcake. As with
13 the standalone evaporator operations, after the concentrated supernates had cooled and the
14 saltcake had settled, the pumpable liquids were routed back to the heater tanks for additional
15 volume reduction.

16 Supernates from B, BX, BY, and C farm tanks were moved through the in-tank heater systems.
17 These in-tank heater systems were used to fill available tank space in the B, BX, and BY tank
18 farms with saltcake. The available space in C tank farm was not filled with saltcake because that
19 farm was geographically too far away from in-tank heater systems. The T Evaporator in the
20 200 West Area was also restarted in 1965 to fill available tank space in the T, TX, and TY tank
21 farms with saltcake.

B9.5   REDUCED PRESSURE EVAPORATORS

23 The S Evaporator came online in 1973 to fill available tank space in the S, SX, and U tank farms
24 with saltcake. The A Evaporator came online in the 200 East Area in 1977. The A and
25 S Evaporators operated under reduced pressure and are sometimes referred to as "crystallizers"
26 rather than "evaporators." The A Evaporator is still in operation. The S Evaporator was placed
27 in standby mode in the early 1980s.

B9.6   IN-PLANT EVAPORATORS

29 An evaporator facility was constructed as part of the REDOX facility (S Plant). During plant
30 operations, the volumes of high-level wastes were routinely reduced by a factor of 2 before being
31 routed to the underground storage tanks. After the S Plant was put on standby in 1967, the
32 REDOX evaporator was used to concentrate tank wastes. An evaporator facility was constructed
33 in B Plant (cell 23) as part of the Isotope Recovery Program that operated from 1967 through
34 1978. The primary function of the B Plant evaporator was to concentrate process waste streams
35 prior to their transfer to the underground storage tanks. However, in 1967, prior to startup of the
36 isotope recovery program, the B Plant evaporator was used to concentrate tank waste so as to
37 free up tank space to support PUREX Plant operations.

B9.7   SUMMARY OF TANK WASTE VOLUME REDUCTIONS

Since the purpose of all of the evaporator operations was the reduction in liquid waste volumes
and thus recovery of tank space, these volumes were documented in various monthly reports.
Anderson (1990) reports a total waste volume reduction of 131 million gal, as of
December 31, 1977. Agnew (1997) reports a total waste volume reduction of 162 million gal,
as of 1991; however, Agnew's results are not comprehensive. A breakdown of volume reduction
per facility is shown in Table B-2, and suggests a total waste volume reduction of 238.3 million
gal as of 1991. The total waste volume reduction does not include volume reductions
accomplished during REDOX facility operations or during B Plant isotope recovery operations.

Steam condensate liquids removed from the various waste streams during waste volume
reduction operations were routinely routed to subsurface soil disposal sites (i.e., cribs).

Table B-2. Waste Volume Reductions

Evaporation Operations	Date	Volume Reduction million gal	Reference
242-B	1951 – 1954	7.9	Agnew 1997
242-T	1951 – 1955	9.9	Agnew 1997
REDOX self-concentration	1952 – 1965	8.4	Agnew 1997
PUREX self-concentration	1956 – 1975	61.1	Agnew 1997
In-tank heaters	1965 – 1976	38.1	Agnew 1997
242-T	1965 – 1976	42.2	Agnew 1997
242-S	1972 – 1977	34.6	Agnew 1997
242-A	1976 – 1980	7.4	Agnew 1997
242-S	1977 – 1980	7.0	Agnew 1997
242-A	1981 – 1991	8.1	Agnew 1997
B Plant (cell 23)	1967 – 1968	1.2	Anderson 1990
REDOX Evaporator	1967 – 1972	12.4	Anderson 1990
		Total = 238.3	

PUREX = plutonium-uranium extraction
REDOX = reduction-oxidation (facility)

B9.8   POTENTIAL IMPACTS OF HANFORD TANK WASTE DEWATERING
OPERATIONS

The dewatering of tank waste fulfilled its two main objectives: 1) maximizing the use of
available tank space and 2) solidifying the tank waste to minimize the impacts of loss of integrity
of the underground storage tanks. However, the various waste solidification programs led to the
formation of multiple types of saltcake. Saltcake composition is dependent on both the original
waste composition and vagaries of the impacts of the liquid phase recycling associated with the
dewatering processes. Thus, the Hanford Site evaporator campaigns have affected both the

distribution and composition of key components in the tank waste. Examples of the potential impacts of the long-term dewatering operations at Hanford Site are listed below.

- Large quantities of sulfate, from the bismuth phosphate separation process, are currently concentrated in some of the B and TX tank farm saltcakes. Sulfate is important because it is one of the key glass-limiting components in LAW glass and, as such, could potentially affect the refractory and the operating life of the LAW glass melter.

- In the BY farm, large quantities of chromium have accumulated in tank BY-112, one of the tanks equipped with an in-tank heater for the ITS program. Other tanks in the BY tank farm, such as BY-107 and BY-108, have become chromium depleted because of the flow patterns and waste consolidation effects introduced by the ITS campaign. Chromium is important because it is one of the key glass-limiting components in high-level waste glass. While some forms of chromium, such as hexavalent chromium (Cr^{+6}), are highly soluble, and thus should be distributed to the LAW glass, other forms such as trivalent chromium (Cr^{+3}) are virtually insoluble and will be largely distributed to the high-level waste glass. Because of radiolysis (the production of hydrogen in an intense radiation field), there is some evidence that large quantities of hexavalent chromium have been reduced to trivalent chromium in some of the self-boiling tanks in the S and SX tank farms.

- Waste evaporator campaigns with REDOX waste were specifically controlled to limit the possible formation of a gelatinous aluminum hydroxide precipitate that could potentially plug the transfer lines from the evaporator to the waste storage tanks in the S and SX tank farms. While potential problems with gelatinous precipitates were generally avoided by careful control of the waste composition, there is some evidence that gelatinous precipitates may have formed in many of the waste receiver tanks. Although the evidence is not clear, gelatinous precipitates may be involved, and quite possibly, could be responsible for the slow retrieval rates and insoluble, hard saltcake layers in many of the saltcake waste receiver tanks. Again, this may be another example of the unintended consequences that have arisen from the evaporator campaigns at the Hanford Site.

- There may be high concentrations of sodium-fluorophosphate double salts in the bottom of several of the waste evaporator feed tanks, specifically tanks S-102 and TX-118. This situation may have been created by the practice of recirculating partially concentrated wastes to the evaporator feed tank where the most insoluble salts were most likely to accumulate. Sodium-fluorophosphate is not only relatively insoluble, but in the presence of excess phosphate, could reprecipitate with needle-shaped crystals, creating gelatinous, plug-forming deposits in the waste transfer lines.

Therefore, it is likely there are a number of unintended consequences of the waste dewatering campaigns that will have to be addressed during retrieval and tank closure activities.

B10.0 SINGLE-SHELL TANK LEAKS

Currently, 67 of the 149 SSTs that were used to store radioactive wastes from the various Hanford Site chemical processing facilities are listed as "confirmed or suspected leakers" (Hanlon 2005). Over the past decade, there has been a significant effort by the Hanford Site Tank Farm Vadose Zone Program to better understand and quantify vadose zone contamination in and around the SSTs. A major focus of the program has been to quantify the inventories of chemicals and radionuclides that were intentionally or accidentally discharged to the vadose zone from SSTs. Inventory values are determined from waste volume estimates and estimates of the composition of the fluids reaching the soil column. Current best estimates of waste loss volumes are listed in Table B-3. Inventory estimates based on these waste volume losses are available (Corbin et al. 2005).

Table B-3 contains revised leak information for the 67 tanks currently listed as "confirmed or suspected leakers" plus tank C-105, leading to the 68 tanks listed in the table. Although tank C-105 is currently listed as a sound tank, gamma activity in one drywell suggested a potential loss of tank supernate sometime during its operational lifetime (Wood et al. 2003). Additional field investigations were completed and data evaluation is underway. The most likely source of gamma contamination in the drywell near tank C-105 was a well-documented waste loss from a pipe connecting tank C-104 to tank C-105 (Field and Jones 2005).

Table B-3. Revised Single-Shell Tank Leak Volume Estimates ^a (3 pages)

Tank/Ancillary Equipment	UPR	March 2005 [b] Leak Volume gal	Revised Leak Volume gal	Waste Composition Year [c]	Group [d]
A-103	NA	5,500	5,500	1987	2
A-104	UPR-200-E-125	500 – 2,500	2,000	1975	2
A-105	UPR-200-E-126	10,000 to 277,000	1,000	1965	3
AX-102	NA	3,000	3,000	1975	2
AX-104	NA	— [e]	No basis for estimate		4
B-101	NA	— [e]	No basis for estimate		4
B-103	NA	— [e]	No basis for estimate		4
B-105	NA	— [e]	No basis for estimate		4
B-107	UPR-200-E-127	8,000	14,000	1965	1
B-110	UPR-200-E-128	10,000	10,000	1969	2
B-111	NA	— [e]	No basis for estimate		4
B-112	NA	2,000	2,000		2
B-201	UPR-200-E-129	1,200	1,200	1965	2
B-203	UPR-200-E-130	300	300	1965	2
B-204	NA	400	400	1965	2
BX-101	UPR-200-E-131	— [e]	4,000	1972	1
BX-102	UPR-200-E-132 UPR-200-E-5	70,000	91,600	1951	1
BX-108	UPR-200-E-133	2,500	2,500	1972	2
BX-110	NA	— [e]	No basis for estimate		4
BX-111	NA	— [e]	No basis for estimate		4
BY-103	UPR-200-E-134	<5,000	See [d-3]	1973	3
BY-105	NA	— [e]	No basis for estimate		4
BY-106	NA	— [e]	No basis for estimate		4
BY-107	NA	15,100	See [d-3]		3
BY-108	UPR-200-E-135	<5,000	See [d-3]	1974	3
C-101	UPR-200-E-136	20,000	1,000	1968	3
C-105	UPR-200-E-16	Not listed	1,000	1972	1
C-110	NA	2,000	2,000	1969	2
C-111	NA	5,500	5,500	1968	2
C-201	NA	550	550	1965	2
C-202	NA	450	450	1965	2

Table B-3. Revised Single-Shell Tank Leak Volume Estimates [a] (3 pages)

Tank/Ancillary Equipment	UPR	March 2005 [b] Leak Volume gal	Revised Leak Volume gal	Waste Composition Year [c]	Group [d]
C-203	UPR-200-E-137	400	400	1957	2
C-204	NA	350	350	1957	2
S-104	NA	24,000	24,000	1965	1
SX-104	NA	6,000	6,000	1988	2
SX-107	UPR-200-W-140	<5,000	15,000	1963	1
SX-108	UPR-200-W-141	2,400 – 35,000	35,000	1966	1
SX-109	UPR-200-W-142	<10,000	2,000	1966	1
SX-110	NA	5,500	1,000	1976	3
SX-111	UPR-200-W-143	500	500	1974	2
SX-112	UPR-200-W-144	30,000	1,000	1968	3
SX-113	UPR-200-W-145	15,000	15,000	1958	1
SX-114	NA	— [e]	No basis for estimate		4
SX-115	UPR-200-W-146	50,000	50,000	1965	1
T-101	NA	7,500	10,000	1969	1
T-103	UPR-200-W-147	<1,000	3,000	1973	1
T-106	UPR-200-W-148	115,000	115,000	1973	1
T-107	NA	— [e]	No basis for estimate		4
T-108	NA	<1,000	1,000	1974	2
T-109	NA	<1,000	1,000	1974	2
T-111	NA	<1,000	1,000	1971	2
TX-105	NA	— [e]	No basis for estimate		4
TX-107	UPR-200-W-149	2,500	8,000	1977	1
TX-110	NA	— [e]	No basis for estimate		4
TX-113	NA	— [e]	No basis for estimate		4
TX-114	NA	— [e]	No basis for estimate		4
TX-115	NA	— [e]	No basis for estimate		4
TX-116	NA	— [e]	No basis for estimate		4
TX-117	NA	— [e]	No basis for estimate		4
TY-101	NA	<1,000	1,000	1973	2
TY-103	UPR-200-W-150	3,000	3,000	1971	1
TY-104	UPR-200-W-151	1,400	1,400	1981	2
TY-105	UPR-200-W-152	35,000	35,000	1957	1

Table B-3. Revised Single-Shell Tank Leak Volume Estimates [a] (3 pages)

Tank/Ancillary Equipment	UPR	March 2005 [b] Leak Volume gal	Revised Leak Volume gal	Waste Composition Year [c]	Group [d]
TY-106	UPR-200-W-153	20,000	20,000	1959	1
U-101	UPR-200-W-154	30,000	5,000	1959	3
U-104	UPR-200-W-155	55,000	55,000	1956	1
U-110	UPR-200-W-156	5,000 – 8,100	6,500	1975	1
U-112	UPR-200-W-157	8,500	8,500	1967	1

[a] There is considerable uncertainty regarding the leak date for many of the SST leaks listed. In general, the leak dates for larger waste loss events are reasonably well known. However, for the smaller waste loss events (i.e., <3,000 gal) many of the leak dates are highly uncertain. The leak dates for tanks SX-111, T-108, and TY-104 are leak confirmation dates identified in Naiknimbalkar (2005) and differ from those used in the SIM (Corbin et al. 2005) as of July 2005. The basis for dates used for these three tanks are discussed in Corbin et al. (2005).

[b] Naiknimbalkar (2005).

[c] Year used in SIM to estimate tank waste composition when a leak started.

[d] Tank leak estimates were placed in one of four groups for uncertainty estimates to be defined and used in the SIM:

    1. Well known and documented
    2. Small leaks, no change in leak volume estimates
    3. No evidence of higher leak volume in vadose zone
    4. No basis for leak volume estimate.

[e] The leak volume estimates in Naiknimbalkar (2005) for these tanks were based on an assumption that their cumulative leakage is approximately the same as for 18 of the 24 tanks where leak volumes were determined by decreases in liquid-level. SSTs SX-110 and T-106 were considered atypical and were not included. SSTs B-201, B-203, B-204, and C-203, also excluded, are small diameter 200-Series tanks. The 18 tank leak estimates that were included in the estimate were SSTs A-103, AX-102, B-107, B-110, BY-107, C-101, C-111, S-104, SX-104, SX-109, T-103, T-108, T-109, T-111, TY-101, TY-104, U-110, and U-112 (Baumhardt 1989). The total liquid loss assumed for the 18 tanks was 150,000 gal, an average of approximately 8,000 gal/tank.

[f] Tank leak estimates for BY tank farm are combined in a total tank farm vadose estimate of 1,160 Ci of cesium-137. The estimate is based on 1996 measurements. Volume estimates were derived from this using the SIM Corbin et al. (2005) and distributed between SSTs BY-103, BY-107, and BY-108.

NA   = not applicable

SIM   = Soil Inventory Model

UPR = unplanned release

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APPENDIX C
INVENTORY INPUTS TO PERFORMANCE ASSESSMENT
MODELING

CONTENTS

TABLES

TERMS

HTWOS   Hanford Tank Waste Operations Simulator
MUST   miscellaneous underground storage tank
WMA   waste management area

C1.0   INTRODUCTION

This appendix provides tables showing inventory for the following major risk-driving contaminants per each identified source term within each waste management area (WMA):

- Tritium
- Carbon-14
- Iodine-129
- Technetium-99
- Chromium
- Fluorine
- Nitrite
- Nitrate
- Uranium.

Additionally, when source terms align parallel to the groundwater flow, a summation of the inventory per source type (e.g., residuals, past leaks) is also provided. For contaminants other than those provided in these tables, Microsoft Excel[1] spreadsheets are provided. The data for these spreadsheets were taken from the following documentation:

- Residual waste in single-shell tanks:

    - *Hanford Tank Waste Operations Simulator Model Data Package for the Development Run for the Refined Target Case* (Kirkbride et al. 2005).

    - Spreadsheet file: AppendixC_TankResiduals.xls.

- Past tank leaks and unplanned releases:

    - Volume of material released: *Tank Farm Vadose Zone Contamination Volume Estimates* (Field and Jones 2005)

    - Inventory of chemicals associated with volume of material released: *Hanford Soil Inventory Model, Rev. 1* (Corbin et al. 2005)

    - Spreadsheet file: AppendixC_PastReleases.xls.

- Ancillary equipment:

    - Remaining in pipelines: *Residual Waste Inventories in the Plugged and Abandoned Pipelines at the Hanford Site* (Lambert 2005)

    - Remaining in miscellaneous underground storage tanks (MUST): For these tanks, it was assumed that MUSTs would be retrieved equivalent to a volume remaining in the 200-Series tanks. For example, if the volume of a MUST was 27,500 gal (half the size of a 200-Series tank), the volume of the residual remaining in the tank would be 15 ft^3, which is half the retrieval goal of a 200-Series tank. Since no inventory numbers are available for the MUSTs, it was assumed the waste material would be an average of what would be left in the single-shell tanks within the WMA (Kirkbride et al. 2005).

    - Spreadsheet file: AppendixC_AncEquipResid.xls.

- Potential retrieval leak:

    - Kirkbride et al. (2005) provides the concentration of contaminants in the fluid during retrieval operations. The inventory provided in this appendix was calculated for an 8,000-gal retrieval leak for the 100-Series tanks.

    - Spreadsheet file: AppendixC_PotRetrievalLeaks_8K.xls.

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[1] Microsoft and Excel are either registered trademarks or trademarks of Microsoft Corporation in the United States and/or other countries.

1 **C2.0 WASTE MANAGEMENT AREA S-SX INVENTORY**

2 This section provides the inventory of the major risk-driving radionuclides and hazardous
3 chemicals found within WMA S-SX.

4 **C2.1 SINGLE-SHELL TANK RESIDUAL WASTE INVENTORY**

**Table C-1. Summary of Final Tank Residual Inventory Estimates Based on HTWOS [a]
Model Assuming 10.2 kL (360 ft[3]) Remaining in the Tank in S Tank Farm**

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO$_2$ kg	NO$_3$ kg	U kg
S-101	1.68E-02	1.96E-03	8.41E-04	6.77E-02	6.19E+01	1.95E-01	4.14E+01	1.20E+02	7.34E+01
S-102	1.46E-02	4.32E-02	1.84E-04	5.87E-01	5.90E+02	9.46E-01	9.52E+01	4.74E+02	7.03E+02
S-103	1.21E-02	3.85E-03	1.42E-04	7.10E-01	7.07E+02	6.65E-01	7.22E+01	3.16E+02	1.14E+02
Sum of rows S101-S103	4.35E-02	4.90E-02	1.17E-03	1.36E+00	1.36E+03	1.81E+00	2.09E+02	9.10E+02	8.90E+02
S-104	3.70E-03	1.02E-03	2.84E-04	2.80E-02	7.54E+00	1.52E-01	1.48E+01	1.38E+02	4.02E+01
S-105	1.85E-02	5.16E-03	1.71E-04	2.15E+00	3.26E+02	5.65E-01	1.50E+01	7.08E+02	1.12E+02
S-106	2.47E-02	6.89E-03	2.20E-04	7.27E-01	7.26E+02	3.19E+01	6.69E+01	9.09E+02	3.51E+01
Sum of rows S104-S106	4.69E-02	1.31E-02	6.75E-04	2.90E+00	1.06E+03	3.26E+01	9.67E+01	1.76E+03	1.87E+02
S-107	3.08E-02	8.88E-04	8.97E-05	9.77E-03	2.26E+01	8.95E+00	1.54E+01	2.92E+01	6.99E+01
S-108	2.54E-02	7.03E-03	2.12E-04	7.00E-01	6.66E+02	1.07E+01	1.05E+02	7.91E+02	8.42E+01
S-109	2.37E-02	6.25E-02	2.00E-04	1.21E+00	4.12E+02	4.67E-01	2.01E+01	1.63E+03	6.31E+01
Sum of rows S107-S109	7.99E-02	7.04E-02	5.02E-04	1.92E+00	1.10E+03	2.01E+01	1.40E+02	2.45E+03	2.17E+02
S-110	2.53E-02	4.80E-03	1.59E-04	2.11E-01	9.69E+01	2.53E+00	3.83E+01	6.52E+02	7.87E+01
S-111	2.96E-02	1.10E-02	2.22E-04	3.28E-01	1.60E+02	3.22E+00	8.37E+01	5.97E+02	7.22E+00
S-112	1.27E-03	7.06E-02	1.05E-03	4.84E-03	1.82E+01	2.53E+00	3.54E+00	1.34E+01	2.36E+02
Sum of rows S110-S112	5.62E-02	8.64E-02	1.43E-03	5.44E-01	2.75E+02	8.28E+00	1.25E+02	1.26E+03	3.22E+02

[a] Kirkbride et al. (2005)

HTWOS = Hanford Tank Waste Operations Simulator

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Table C-2. Summary of Final Tank Residual Inventory Estimates Based on HTWOS [a] Model Assuming 10.2 kL (360 ft^3) Remaining in the Tank in SX Tank Farm

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO$_2$ kg	NO$_3$ kg	U kg
SX-101	5.10E-02	4.83E-03	1.86E-04	1.93E-01	1.18E+03	2.61E+00	7.67E+01	9.32E+02	2.54E+02
SX-102	2.42E-02	6.04E-03	2.22E-04	9.80E-01	6.35E+02	4.03E+00	2.04E+02	6.44E+02	7.28E+01
SX-103	2.71E-02	5.71E-03	1.88E-04	2.09E-01	1.79E+02	1.52E+00	1.24E+02	6.93E+02	2.04E+01
Sum of rows SX101-SX103	1.02E-01	1.66E-02	5.96E-04	1.38E+00	1.99E+03	8.16E+00	4.05E+02	2.27E+03	3.47E+02
SX-104	2.01E-02	2.80E-03	9.23E-05	1.36E-01	6.69E+01	1.42E+00	5.13E+01	3.30E+02	4.22E+01
SX-105	2.58E-02	6.05E-03	2.15E-04	4.27E-01	2.76E+02	3.34E+00	1.90E+02	5.95E+02	6.63E+01
SX-106	2.97E-02	3.22E-02	2.89E-04	7.33E-01	5.60E+02	1.44E+00	2.22E+02	6.99E+02	5.16E+01
Sum of rows SX104-SX106	7.56E-02	4.11E-02	5.97E-04	1.30E+00	9.03E+02	6.20E+00	4.63E+02	1.62E+03	1.60E+02
SX-107	7.34E-03	6.99E-04	1.65E-04	1.52E-02	8.33E+00	7.78E-02	1.42E+01	6.96E+01	3.71E+01
SX-108	4.99E-02	4.16E-03	5.08E-04	7.16E-02	8.78E+01	1.40E+00	5.07E+01	1.21E+03	1.50E+02
SX-109	2.41E-02	1.94E-03	6.88E-04	6.01E-02	1.78E+01	4.00E-02	1.69E+01	2.82E+02	2.11E+01
Sum of rows SX107-SX109	8.14E-02	6.79E-03	1.36E-03	1.47E-01	1.14E+02	1.52E+00	8.18E+01	1.56E+03	2.08E+02
SX-110	2.16E-02	1.53E-03	4.96E-05	4.70E-02	1.58E+01	9.56E-02	2.27E+01	2.42E+02	3.63E+01
SX-111	1.22E-02	1.38E-03	2.70E-05	2.58E-02	1.11E+01	8.17E-02	1.82E+01	1.18E+02	3.94E+01
SX-112	9.97E-03	1.20E-03	2.14E-05	2.10E-02	1.01E+01	8.38E-02	1.68E+01	9.55E+01	4.03E+01
Sum of rows SX110-SX112	4.37E-02	4.11E-03	9.79E-05	9.38E-02	3.70E+01	2.61E-01	5.78E+01	4.56E+02	1.16E+02
SX-113	4.24E-04	2.23E-04	7.70E-07	7.52E-04	4.38E-01	7.04E-03	1.10E+00	3.90E+00	9.70E-01
SX-114	2.60E-02	1.93E-03	5.92E-04	5.87E-02	1.64E+01	4.86E-02	2.49E+01	2.57E+02	2.46E+01
SX-115	4.56E-02	7.21E-04	4.21E-05	4.25E-02	2.92E+02	0.00E+00	1.71E+00	2.71E-08	8.99E+01
Sum of rows SX113-SX115	7.20E-02	2.87E-03	6.35E-04	1.02E-01	3.09E+02	5.57E-02	2.77E+01	2.60E+02	1.15E+02

[a] Kirkbride et al. (2005)

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1   **C2.2   PAST TANK LEAKS AND UNPLANNED RELEASES INVENTORY**

**Table C-3. Summary of Past Leak Inventory Estimates within S Tank Farm
Based on Volume Estimates [a] and Soil Inventory Model [b]**

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO$_2$ kg	NO$_3$ kg	NH$_3$ kg	Butanol kg	U kg
S-104	2.54E+01	1.73E-01	5.57E-05	3.95E-02	1.45E+01	0.00E+00	1.20E+03	4.99E+03	6.87E-02	0.00E+00	2.82E+00
Sum of rows S104-S106	2.54E+01	1.73E-01	5.57E-05	3.95E-02	1.45E+01	0.00E+00	1.20E+03	4.99E+03	6.87E-02	0.00E+00	2.82E+00

[a] Field and Jones (2005)

[b] Corbin et al. (2005)

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**Table C-4. Summary of Past Leak Inventory Estimates within SX Tank Farm
Based on Volume Estimates [a] and Soil Inventory Model [b]**

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO$_2$ kg	NO$_3$ kg	NH$_3$ kg	Butanol kg	U kg
SX-104	6.18E-01	1.46E-01	4.39E-03	4.51E+00	6.17E+01	3.62E+01	2.46E+03	4.65E+03	3.66E+01	2.41E+01	6.68E-01
Sum of rows SX104-SX106	6.18E-01	1.46E-01	4.39E-03	4.51E+00	6.17E+01	3.62E+01	2.46E+03	4.65E+03	3.66E+01	2.41E+01	6.68E-01
SX-107	4.20E+00	1.73E-01	9.28E-03	6.01E+00	1.63E+02	2.27E-01	5.93E+03	1.23E+04	4.21E+01	3.86E+00	1.76E+00
SX-108	9.81E+00	4.04E-01	2.17E-02	1.40E+01	3.81E+02	5.29E-01	1.38E+04	2.87E+04	9.82E+01	9.02E+00	4.11E+00
SX-109	5.61E-01	2.31E-02	1.24E-03	8.01E-01	2.18E+01	3.02E-02	7.90E+02	1.64E+03	5.61E+00	5.15E-01	2.35E-01
Sum of rows SX107-SX109	1.46E+01	6.00E-01	3.22E-02	2.09E+01	5.67E+02	7.86E-01	2.05E+04	4.27E+04	1.46E+02	1.34E+01	6.10E+00
SX-110	8.26E-02	9.65E-03	2.59E-04	3.35E-01	4.43E+00	2.21E+00	1.17E+02	2.86E+02	1.64E+00	3.67E-01	8.66E-02
SX-111	4.86E-02	6.05E-03	1.56E-04	2.28E-01	2.82E+00	7.07E-02	7.05E+01	1.44E+02	7.12E-01	5.10E-02	5.50E-02
SX-112	2.80E-01	1.15E-02	6.19E-04	4.01E-01	1.09E+01	1.51E-02	3.95E+02	8.21E+02	2.80E+00	2.58E-01	1.17E-01
Sum of rows SX110-SX112	4.12E-01	2.72E-02	1.03E-03	9.63E-01	1.82E+01	2.30E+00	5.82E+02	1.25E+03	5.16E+00	6.76E-01	2.59E-01
SX-113	1.56E+00	8.51E-02	2.39E-03	1.50E+00	1.62E+02	5.11E-05	1.99E+03	7.87E+03	9.17E+00	0.00E+00	1.75E+00
SX-115	3.83E+01	3.05E-01	7.04E-03	4.53E+00	2.37E+02	5.51E-03	6.04E+03	1.43E+04	3.41E+01	9.20E-02	5.84E+00
Sum of rows SX113-SX115	3.99E+01	3.90E-01	9.44E-03	6.03E+00	4.00E+02	5.56E-03	8.03E+03	2.21E+04	4.33E+01	9.20E-02	7.59E+00

[a] Field and Jones (2005)

[b] Corbin et al. (2005)

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4   **C2.3   ANCILLARY EQUIPMENT (PIPELINES AND MUST) INVENTORY**

Table C-5. Summary of Final Pipeline Inventory Estimates for S Tank Farm [a]

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO$_2$ kg	NO$_3$ kg	U kg
Pipeline	2.16E-03	1.59E-04	1.65E-05	1.77E-02	2.23E-01	6.54E-03	8.46E+00	1.71E+01	2.47E-03

[a] Lambert (2005)

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**Table C-6. Summary of Extrapolated MUST Inventory for
Waste Management Area S-SX**

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO$_2$ kg	NO$_3$ kg	U kg
241-S-302 placed in tank row SX101-SX103	6.70E-04	1.28E-04	5.87E-06	5.40E-03	5.99E+00	2.89E-02	1.85E+00	1.10E+01	1.69E+00

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C2.4   POTENTIAL RETRIEVAL LEAKS

**Table C-7. Summary of 8,000-Gallon Retrieval Leak Inventory Estimates Based on
HTWOS [a] Model Estimated Concentration for Retrieval Fluids for
Single-Shell Tanks within S Tank Farm**

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO$_2$ kg	NO$_3$ kg	U kg
S-101	1.18E-01	7.96E-03	5.11E-05	2.98E-01	1.33E+01	1.38E+00	2.91E+02	8.42E+02	3.13E-04
S-102	1.04E-01	2.76E-02	1.30E-03	1.02E+00	1.96E+00	6.71E+00	5.18E+02	2.49E+03	5.51E-01
S-103	8.58E-02	2.35E-02	1.01E-03	1.02E+00	5.19E+00	4.71E+00	5.12E+02	2.24E+03	2.78E-03
Sum of rows S101-S103	3.08E-01	5.90E-02	2.36E-03	2.34E+00	2.05E+01	1.28E+01	1.32E+03	5.58E+03	5.54E-01
S-104	2.60E-02	4.45E-03	1.14E-04	1.26E-01	7.89E+00	1.07E+00	1.04E+02	9.71E+02	9.20E-03
S-105	1.31E-01	3.66E-02	1.22E-03	1.24E+00	2.47E+00	4.01E+00	1.06E+02	5.03E+03	1.26E-03
S-106	1.77E-01	4.94E-02	1.58E-03	1.61E+00	2.14E+01	3.42E+01	4.51E+02	5.72E+03	2.46E-01
Sum of rows S104-S106	3.34E-01	9.04E-02	2.91E-03	2.98E+00	3.17E+01	3.93E+01	6.62E+02	1.17E+04	2.57E-01
S-107	2.16E-01	3.56E-03	2.21E-05	4.34E-02	5.51E+00	6.28E+01	1.08E+02	2.05E+02	0.00E+00
S-108	1.82E-01	5.04E-02	1.52E-03	1.55E+00	1.59E+01	1.15E+01	7.04E+02	4.96E+03	5.87E-01
S-109	1.70E-01	4.39E-02	1.43E-03	1.46E+00	6.96E+00	3.34E+00	1.04E+02	8.75E+03	5.92E-02
Sum of rows S107-S109	5.68E-01	9.78E-02	2.97E-03	3.06E+00	2.83E+01	7.77E+01	9.15E+02	1.39E+04	6.46E-01
S-110	1.80E-01	3.41E-02	1.13E-03	1.15E+00	2.89E+01	7.19E+00	2.68E+02	4.47E+03	2.06E+00
S-111	2.13E-01	4.02E-02	1.60E-03	1.61E+00	2.60E+01	2.31E+01	5.71E+02	4.02E+03	1.04E+00
S-112	9.13E-03	5.06E-01	7.55E-03	1.00E-02	4.22E-01	2.68E+00	2.37E+01	8.39E+01	1.63E+00
Sum of rows S110-S112	4.01E-01	5.80E-01	1.03E-02	2.78E+00	5.53E+01	3.30E+01	8.62E+02	8.58E+03	4.73E+00

[a] Kirkbride et al. (2005)

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Table C-8. Summary of 8,000-Gallon Retrieval Leak Inventory Estimates Based on HTWOS [a] Model Estimated Concentration for Retrieval Fluids for Single-Shell Tanks within SX Tank Farm

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO$_2$ kg	NO$_3$ kg	U kg
SX-101	3.70E-01	3.50E-02	1.35E-03	1.12E+00	5.90E+01	2.95E+00	4.45E+02	5.32E+03	1.31E+00
SX-102	1.73E-01	4.34E-02	1.59E-03	1.56E+00	2.36E+01	5.29E+00	1.40E+03	4.17E+03	6.41E-01
SX-103	1.93E-01	4.08E-02	1.34E-03	1.36E+00	7.79E+00	4.41E+00	8.72E+02	4.78E+03	5.53E-01
Sum of rows SX101-SX103	7.37E-01	1.19E-01	4.28E-03	4.04E+00	9.04E+01	1.26E+01	2.72E+03	1.43E+04	2.50E+00
SX-104	1.42E-01	1.98E-02	6.54E-04	5.89E-01	1.44E+01	3.97E+00	3.57E+02	2.25E+03	1.09E+00
SX-105	1.86E-01	4.36E-02	1.55E-03	1.57E+00	1.37E+01	5.65E+00	1.32E+03	3.97E+03	7.99E-01
SX-106	2.13E-01	5.75E-02	2.08E-03	2.08E+00	7.26E+00	1.03E+01	1.39E+03	4.54E+03	1.48E-01
Sum of rows SX104-SX106	5.41E-01	1.21E-01	4.28E-03	4.23E+00	3.54E+01	1.99E+01	3.07E+03	1.08E+04	2.03E+00
SX-107	5.16E-02	2.89E-03	5.13E-05	6.52E-02	8.11E+00	5.46E-01	9.95E+01	4.89E+02	1.38E-01
SX-108	3.59E-01	2.24E-02	2.76E-04	3.94E-01	1.58E+02	1.01E+01	3.65E+02	8.68E+03	0.00E+00
SX-109	1.70E-01	7.85E-03	1.73E-04	2.56E-01	2.49E+01	2.82E-01	1.19E+02	1.99E+03	7.90E-02
Sum of rows SX107-SX109	5.81E-01	3.31E-02	5.01E-04	7.15E-01	1.91E+02	1.09E+01	5.84E+02	1.12E+04	2.17E-01
SX-110	1.52E-01	7.55E-03	3.50E-04	2.17E-01	2.30E+01	6.74E-01	1.60E+02	1.71E+03	3.60E-08
SX-111	8.58E-02	4.32E-03	1.90E-04	1.18E-01	1.23E+01	5.74E-01	1.28E+02	8.29E+02	5.98E-03
SX-112	7.01E-02	3.65E-03	1.50E-04	9.30E-02	1.05E+01	5.89E-01	1.18E+02	6.71E+02	4.07E-03
Sum of rows SX110-SX112	3.08E-01	1.55E-02	6.89E-04	4.28E-01	4.57E+01	1.84E+00	4.07E+02	3.21E+03	1.00E-02
SX-113	2.97E-03	1.22E-04	5.40E-06	3.27E-03	3.06E-03	4.93E-02	7.72E+00	2.74E+01	1.08E-02
SX-114	1.83E-01	8.57E-03	2.42E-04	2.77E-01	2.30E+01	3.42E-01	1.76E+02	1.81E+03	1.11E-01
SX-115	3.21E-01	2.21E-03	2.96E-04	1.85E-01	2.93E+02	0.00E+00	1.20E+01	1.91E-07	9.58E-11
Sum of rows SX113-SX115	5.07E-01	1.09E-02	5.44E-04	4.66E-01	3.16E+02	3.92E-01	1.95E+02	1.83E+03	1.22E-01

[a] Kirkbride et al. (2005)

1 **C3.0 WASTE MANAGEMENT AREA T INVENTORY**

2 This section provides the inventory of the major risk-driving radionuclides and hazardous
3 chemicals found within WMA T.

4 **C3.1 SINGLE-SHELL TANK RESIDUAL WASTE INVENTORY**

Table C-9. Summary of Final Tank Residual Inventory Estimates Based on HTWOS[a] Model Assuming 10.2 kL (360 ft³) Remaining in the Tank in Waste Management Area T

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO₂ kg	NO₃ kg	U kg
T-101	1.23E-01	3.85E-03	1.50E-04	3.42E-01	9.80E+00	3.55E+00	1.04E+02	8.40E+02	1.64E+02
T-102	6.24E-03	2.56E-02	1.20E-04	6.84E-02	9.83E-01	2.02E-01	1.58E+01	6.46E+01	2.27E+02
T-103	1.67E-02	9.10E-04	8.51E-05	1.62E-02	1.38E+00	7.36E-01	1.90E+01	4.38E+01	1.40E+02
Sum of rows T101-T103	1.46E-01	3.04E-02	3.55E-04	4.27E-01	1.22E+01	4.49E+00	1.39E+02	9.49E+02	5.31E+02
T-104	1.02E-03	1.87E-04	1.38E-06	2.64E-03	2.21E+01	2.03E+02	1.78E+01	2.43E+02	2.53E+01
T-105	3.29E-03	3.36E-03	6.49E-08	5.50E-01	2.11E+01	1.05E+00	6.50E+01	3.07E+02	3.77E+01
T-106	5.26E-02	4.76E-04	5.24E-07	6.64E-02	5.05E+00	8.31E+00	2.83E+01	1.88E+02	5.43E+01
Sum of rows T104-T106	5.69E-02	4.03E-03	1.97E-06	6.19E-01	4.83E+01	2.13E+02	1.11E+02	7.39E+02	1.17E+02
T-107	3.49E-03	1.14E-03	3.70E-05	3.00E-01	5.34E+00	1.36E+02	7.03E+01	4.48E+02	1.71E+02
T-108	3.14E-03	3.88E-04	3.56E-06	2.26E-02	3.58E+00	1.02E+02	2.00E+01	7.63E+02	1.48E+01
T-109	3.59E-03	3.11E-04	2.19E-05	1.45E-02	2.58E+00	4.33E+02	5.73E+00	1.74E+02	4.36E+01
T-201	5.37E-11	1.42E-07	5.68E-12	1.28E-08	4.82E+00	5.56E+00	4.59E-01	5.00E+01	7.47E-06
T-202	4.26E-09	2.08E-07	0.00E+00	1.91E-08	3.40E+00	6.36E+00	6.75E-01	6.50E+01	1.13E-01
Sum of rows T107-T109 and T201-T202	1.02E-02	1.84E-03	6.24E-05	3.38E-01	1.97E+01	6.82E+02	9.72E+01	1.50E+03	2.30E+02
T-110	1.05E-06	2.53E-05	2.49E-09	1.10E-05	4.99E+01	2.03E+01	2.34E+02	8.32E+02	9.79E-01
T-111	5.19E-07	1.17E-05	9.98E-10	1.06E-01	5.55E+01	6.11E+01	9.89E+00	6.90E+02	1.27E+02
T-112	6.00E-07	1.15E-05	8.75E-10	4.70E-06	5.16E+01	2.99E+00	2.39E+02	1.28E+02	1.73E+01
T-203	4.24E-09	2.07E-07	0.00E+00	1.90E-08	3.78E+00	6.60E+00	3.56E-01	6.25E+01	3.44E-03
T-204	4.39E-09	2.15E-07	0.00E+00	1.97E-08	4.85E+00	6.42E+00	3.06E-01	5.96E+01	1.19E-03
Sum of rows T110-T112 and T203-T204	2.17E-06	4.90E-05	4.36E-09	1.06E-01	1.66E+02	9.74E+01	4.83E+02	1.77E+03	1.45E+02

[a] Kirkbride et al. (2005)

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1   **C3.2   PAST TANK LEAKS AND UNPLANNED RELEASES INVENTORY**

Table C-10. Summary of Past Leak Inventory Estimates within Waste Management Area T Based on Volume Estimates [a] and Soil Inventory Model [b]

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO_2 kg	NO_3 kg	NH_3 kg	Butanol kg	U kg
T-101	6.12E+00	5.84E-02	4.68E-04	3.04E-01	1.26E+01	1.20E-02	1.46E+03	2.15E+03	2.11E+00	1.88E-01	9.52E-01
T-103	3.92E-01	2.58E-02	7.09E-04	9.39E-01	1.24E+01	2.29E-01	3.42E+02	7.26E+02	3.05E+00	6.59E-02	2.97E-01
Sum of rows T101-T103	6.51E+00	8.42E-02	1.18E-03	1.24E+00	2.50E+01	2.41E-01	1.80E+03	2.88E+03	5.16E+00	2.54E-01	1.25E+00
T-106	1.54E+01	1.05E+00	2.46E-02	3.74E+01	5.04E+02	3.12E+01	1.33E+04	2.95E+04	1.52E+02	3.46E+00	1.19E+01
Sum of rows T104-T106	1.54E+01	1.05E+00	2.46E-02	3.74E+01	5.04E+02	3.12E+01	1.33E+04	2.95E+04	1.52E+02	3.46E+00	1.19E+01
T-108	1.83E-02	1.02E-03	1.71E-05	1.23E-02	1.66E+00	1.22E+01	3.16E+01	8.05E+02	5.38E+00	2.30E-03	1.18E-01
T-109	7.86E-02	8.72E-03	2.37E-04	3.06E-01	4.24E+00	2.91E+00	1.10E+02	3.67E+02	2.33E+00	2.59E-01	9.77E-02
Sum of rows T107-T109 and T201-T202	9.69E-02	9.75E-03	2.54E-04	3.18E-01	5.89E+00	1.51E+01	1.41E+02	1.17E+03	7.72E+00	2.62E-01	2.15E-01
T-111	4.33E-06	1.37E-05	1.23E-10	7.40E-06	5.96E-01	6.48E+00	7.63E-03	1.19E+02	1.21E-08	0.00E+00	1.21E-02
Sum of rows T110-T112 and T203-T204	4.33E-06	1.37E-05	1.23E-10	7.40E-06	5.96E-01	6.48E+00	7.63E-03	1.19E+02	1.21E-08	0.00E+00	1.21E-02

[a] Field and Jones (2005)

[b] Corbin et al. (2005)

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3   **C3.3   ANCILLARY EQUIPMENT (PIPELINES AND MUST) INVENTORY**

Table C-11. Summary of Extrapolated MUST Inventory for Waste Management Area T

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO_2 kg	NO_3 kg	U kg
241-T-301B placed in tank row T111–T112	9.41E-04	1.60E-04	1.85E-06	6.58E-03	1.08E+00	4.40E+00	3.67E+00	2.19E+01	4.52E+00

4

1 **C3.4 POTENTIAL RETRIEVAL LEAKS**

**Table C-12. Summary of 8,000-Gallon Retrieval Leak Inventory Estimates Based
on HTWOS [a] Model Estimated Concentration for Retrieval Fluids for
Single-Shell Tanks within Waste Management Area T**

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO$_2$ kg	NO$_3$ kg	U kg
T-101	8.73E-01	2.66E-02	1.06E-03	8.56E-01	1.06E+01	2.41E+01	4.33E+02	3.31E+03	2.20E-02
T-102	4.38E-02	1.75E-01	8.44E-04	2.91E-01	6.38E+00	1.38E+00	7.36E+01	3.44E+02	5.26E-02
T-103	1.18E-01	6.29E-03	5.98E-04	9.83E-02	2.17E+00	4.93E+00	6.98E+01	1.83E+02	3.48E-02
Sum of rows T101-T103	1.03E+00	2.08E-01	2.51E-03	1.25E+00	1.92E+01	3.04E+01	5.76E+02	3.83E+03	1.09E-01
T-104	7.18E-03	1.32E-03	9.77E-06	1.87E-02	4.27E+00	4.87E+01	1.26E+02	1.72E+03	2.15E-01
T-105	2.32E-02	2.18E-03	4.58E-07	5.53E-01	7.55E+00	6.91E+00	4.59E+02	1.02E+02	2.25E+00
T-106	3.70E-01	3.26E-03	3.69E-06	9.24E-02	2.17E+00	5.58E+01	8.19E+01	6.85E+02	5.73E-03
Sum of rows T104-T106	4.01E-01	6.76E-03	1.39E-05	6.64E-01	1.40E+01	1.11E+02	6.66E+02	2.51E+03	2.47E+00
T-107	2.51E-02	8.16E-03	2.66E-04	2.16E+00	1.14E+01	3.01E+02	5.05E+02	3.22E+03	1.35E+01
T-108	2.23E-02	2.76E-03	2.53E-05	1.61E-01	3.72E+00	1.16E+02	1.42E+02	5.42E+03	1.31E+01
T-109	2.58E-02	1.77E-03	2.48E-05	1.81E-02	1.83E+00	2.45E+02	3.74E+01	9.95E+02	3.08E-01
Sum of rows T107-T109	7.32E-02	1.27E-02	3.16E-04	2.34E+00	1.69E+01	6.62E+02	6.85E+02	9.64E+03	2.69E+01
T-110	7.40E-06	3.35E-05	4.19E-09	2.05E-05	1.33E+00	3.69E+01	4.43E+02	1.88E+03	1.11E-03
T-111	3.67E-06	1.88E-05	1.99E-09	2.34E-01	5.98E+00	1.32E+02	2.23E+01	1.83E+03	0.00E+00
T-112	4.22E-06	1.74E-05	8.03E-10	9.62E-06	4.23E+00	3.43E+00	2.90E+02	1.87E+02	9.97E-01
Sum of rows T110-T112	1.53E-05	6.96E-05	6.98E-09	2.34E-01	1.15E+01	1.73E+02	7.56E+02	3.89E+03	9.98E-01

[a] Kirkbride et al. (2005)

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1 **C4.0 WASTE MANAGEMENT AREA TX-TY INVENTORY**

2 This section provides the inventory of the major risk-driving radionuclides and hazardous
3 chemicals found within WMA TX-TY.

4 **C4.1 SINGLE-SHELL TANK RESIDUAL WASTE INVENTORY**

**Table C-13. Summary of Final Tank Residual Inventory Estimates Based on HTWOS [a]
Model Assuming 10.2 kL (360 ft[3]) Remaining in the Tank in TX Tank Farm**

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO$_2$ kg	NO$_3$ kg	U kg
TX-101	5.22E-02	5.23E-03	5.96E-04	1.67E-01	5.50E+01	2.39E+00	1.41E+02	6.35E+02	4.10E+02
TX-102	3.32E-02	1.44E-02	2.27E-04	3.51E+00	4.40E+01	6.25E+00	9.20E+01	9.11E+02	3.25E+02
TX-103	3.34E-02	1.49E-02	2.16E-04	3.70E+00	4.52E+01	6.78E+00	8.71E+01	9.28E+02	6.93E+01
TX-104	3.62E-02	6.71E-03	2.02E-04	1.05E+00	6.47E+01	5.79E+00	1.12E+02	1.04E+03	5.01E+01
Sum of rows TX101-TX104	1.55E-01	4.12E-02	1.24E-03	8.42E+00	2.09E+02	2.12E+01	4.32E+02	3.51E+03	8.54E+02
TX-105	3.60E-02	1.45E-02	2.00E-04	3.39E+00	4.31E+01	6.74E+00	7.66E+01	9.49E+02	3.97E+02
TX-106	3.28E-02	5.16E-03	2.08E-04	1.96E-01	6.77E+01	6.10E+00	7.83E+01	8.30E+02	2.26E+02
TX-107	4.76E-02	1.70E-02	2.35E-04	4.76E+00	1.40E+02	4.42E+00	9.39E+01	9.40E+02	7.56E+01
TX-108	3.44E-02	1.37E-02	2.06E-04	3.18E+00	4.03E+01	6.49E+00	8.17E+01	9.29E+02	4.80E+02
Sum of rows TX105-TX108	1.51E-01	5.03E-02	8.49E-04	1.15E+01	2.91E+02	2.38E+01	3.31E+02	3.65E+03	1.18E+03
TX-109	1.86E-03	1.13E-03	2.53E-06	1.20E-01	3.84E+01	5.03E+01	4.79E+01	6.21E+02	1.02E+02
TX-110	3.54E-02	1.40E-02	1.89E-04	3.29E+00	4.36E+01	8.57E+00	7.34E+01	9.52E+02	6.97E+01
TX-111	3.50E-02	1.33E-02	1.83E-04	3.09E+00	4.24E+01	9.49E+00	7.13E+01	9.48E+02	6.97E+01
TX-112	3.51E-02	1.51E-02	1.91E-04	3.66E+00	4.51E+01	7.49E+00	7.38E+01	9.60E+02	7.04E+01
Sum of rows TX109-TX112	1.07E-01	4.35E-02	5.65E-04	1.02E+01	1.70E+02	7.58E+01	2.66E+02	3.48E+03	3.12E+02
TX-113	3.45E-02	1.69E-02	1.76E-04	4.46E+00	9.80E+00	1.18E+01	9.69E+00	1.02E+03	3.71E+01
TX-114	3.34E-02	1.09E-02	1.71E-04	2.25E+00	2.92E+01	8.94E+00	6.38E+01	9.39E+02	4.63E+01
TX-115	3.59E-02	1.53E-02	2.00E-04	3.71E+00	4.57E+01	6.76E+00	7.77E+01	9.61E+02	8.23E+01
Sum of rows TX113-TX115	1.04E-01	4.32E-02	5.47E-04	1.04E+01	8.47E+01	2.75E+01	1.51E+02	2.92E+03	1.66E+02
TX-116	1.61E-02	5.15E-03	1.07E-04	1.54E+00	1.27E+01	7.44E+00	2.99E+01	1.06E+03	4.29E+01
TX-117	2.28E-02	7.92E-03	1.10E-04	1.70E+00	2.23E+01	1.41E+01	4.24E+01	8.97E+02	5.12E+01
TX-118	2.59E-02	1.13E-02	1.87E-04	2.78E+00	1.08E+02	4.14E+01	6.23E+01	2.12E+02	7.38E+01
Sum of rows TX116-TX118	6.48E-02	2.43E-02	4.04E-04	6.02E+00	1.43E+02	6.30E+01	1.35E+02	2.17E+03	1.68E+02

[a] Kirkbride et al. (2005)

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Table C-14. Summary of Final Tank Residual Inventory Estimates Based on HTWOS [a] Model Assuming 10.2 kL (360 ft³) Remaining in the Tank in TY Tank Farm

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO₂ kg	NO₃ kg	U kg
TY-101	1.80E-04	2.80E-03	5.74E-07	2.87E-02	2.27E+01	1.76E+00	4.78E+00	9.90E+02	1.98E+01
TY-102	1.85E-02	3.34E-03	7.21E-05	9.34E-02	2.73E+00	1.37E+01	5.88E+01	1.20E+03	1.37E+01
Sum of rows TY101-TY102	1.86E-02	6.14E-03	7.27E-05	1.22E-01	2.54E+01	1.54E+01	6.35E+01	2.19E+03	3.36E+01
TY-103	3.80E-03	4.50E-03	2.98E-05	1.38E-01	1.31E+01	1.90E+00	2.30E+01	3.17E+02	4.01E+02
TY-104	4.77E-04	1.99E-02	1.23E-06	1.74E-01	7.97E+00	4.02E+00	2.09E+01	8.13E+01	3.63E+02
Sum of rows TY103-TY104	4.28E-03	2.44E-02	3.11E-05	3.12E-01	2.10E+01	5.92E+00	4.39E+01	3.99E+02	7.65E+02
TY-105	2.40E-03	1.10E-04	4.93E-05	4.71E-02	3.13E+00	9.34E+00	2.24E+01	2.55E+02	1.53E+02
TY-106	1.64E-04	9.83E-06	9.82E-07	1.68E-02	3.86E-01	2.21E-01	4.07E+00	9.94E+01	4.77E+01
Sum of rows TY105-TY106	2.56E-03	1.20E-04	5.03E-05	6.39E-02	3.51E+00	9.56E+00	2.65E+01	3.54E+02	2.00E+02

[a] Kirkbride et al. (2005)

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2 **C4.2   PAST TANK LEAKS AND UNPLANNED RELEASES INVENTORY**

Table C-15. Summary of Past Leak Inventory Estimates within TX Tank Farm Based on Volume Estimates [a] and Soil Inventory Model [b]

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO₂ kg	NO₃ kg	NH₃ kg	Butanol kg	U kg
TX-107	1.12E+00	1.49E-01	4.81E-03	4.37E+00	8.65E+01	7.58E+01	2.50E+03	6.51E+03	6.14E+01	2.12E+01	9.32E-01
UPR-200-W-100	5.88E-02	7.80E-04	1.70E-05	1.96E-03	3.45E+00	3.61E+01	2.23E+01	1.03E+03	1.39E+01	0.00E+00	3.00E-01
Sum of rows TX105-TX108	1.18E+00	1.50E-01	4.82E-03	4.37E+00	8.99E+01	1.12E+02	2.53E+03	7.54E+03	7.53E+01	2.12E+01	1.23E+00

[a] Field and Jones (2005)

[b] Corbin et al. (2005)

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Table C-16. Summary of Past Leak Inventory Estimates within TY Tank Farm Based on Volume Estimates [a] and Soil Inventory Model [b]

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO$_2$ kg	NO$_3$ kg	NH$_3$ kg	Butanol kg	U kg
TY-101	1.51E-01	1.49E-02	5.01E-04	4.45E-01	9.49E+00	7.44E+00	2.66E+02	7.28E+02	6.07E+00	2.03E+00	1.13E-01
Sum of rows TY101-TY102	1.51E-01	1.49E-02	5.01E-04	4.45E-01	9.49E+00	7.44E+00	2.66E+02	7.28E+02	6.07E+00	2.03E+00	1.13E-01
TY-103	4.21E-01	5.60E-02	1.81E-03	1.65E+00	3.25E+01	2.85E+01	9.41E+02	2.44E+03	2.31E+01	7.96E+00	3.52E-01
TY-104	0.00E+00	2.07E-08	2.92E-10	0.00E+00	0.00E+00	7.23E-04	0.00E+00	2.99E-03	0.00E+00	0.00E+00	2.55E-04
Sum of rows TY103-TY104	4.21E-01	5.60E-02	1.81E-03	1.65E+00	3.25E+01	2.85E+01	9.41E+02	2.44E+03	2.31E+01	7.96E+00	3.52E-01
TY-105	8.12E-01	3.15E-02	6.35E-04	4.95E-01	2.20E+01	0.00E+00	9.72E-02	2.41E-04	9.02E-01	9.02E-02	4.11E+00
TY-106	4.64E-01	1.80E-02	3.63E-04	2.83E-01	1.26E+01	0.00E+00	5.55E+02	1.38E+04	5.15E-01	5.15E-02	2.35E+00
Sum of rows TY105-TY106	1.28E+00	4.94E-02	9.97E-04	7.77E-01	3.46E+01	0.00E+00	1.53E+03	3.79E+04	1.42E+00	1.42E-01	6.46E+00

[a] Field and Jones (2005)
[b] Corbin et al. (2005)

C4.3 ANCILLARY EQUIPMENT (PIPELINES AND MUST) INVENTORY

Table C-17. Summary of Final Pipeline Inventory Estimates for TX Tank Farm [a]

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO$_2$ kg	NO$_3$ kg	U kg
Pipeline	7.33E-03	4.00E-04	1.13E-05	7.03E-03	7.63E-01	0.00E+00	9.38E+00	3.70E+01	8.22E-03

[a] Lambert (2005)

Table C-18. Summary of Extrapolated MUST Inventory for Waste Management Area TX-TY

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO$_2$ kg	NO$_3$ kg	U kg
244-TXR vault	3.86E-03	1.34E-03	2.39E-05	3.09E-01	5.95E+00	1.40E+00	8.72E+00	1.04E+02	1.78E+01
241-TX-302A	8.98E-04	3.13E-04	5.56E-06	7.18E-02	1.38E+00	3.26E-01	2.03E+00	2.43E+01	4.13E+00
Sum of rows 244-TXR vault and 241-TX-302A	4.76E-03	1.66E-03	2.95E-05	3.81E-01	7.34E+00	1.73E+00	1.08E+01	1.29E+02	2.19E+01
241-TX-302XB placed in tank row TX101–TX104	8.98E-04	3.13E-04	5.56E-06	7.18E-02	1.38E+00	3.26E-01	2.03E+00	2.43E+01	4.13E+00
241-TY-302B placed in tank row TY101–Y102	9.44E-05	1.14E-04	5.70E-07	1.84E-03	1.85E-01	1.14E-01	4.96E-01	1.09E+01	3.70E+00
241-TY-302A placed in tank row TY105–TY106	1.18E-04	1.42E-04	7.13E-07	2.31E-03	2.31E-01	1.43E-01	6.20E-01	1.36E+01	4.62E+00

1   **C4.4   POTENTIAL RETRIEVAL LEAKS**

Table C-19. Summary of 8,000-Gallon Retrieval Leak Inventory Estimates Based on HTWOS [a] Model Estimated Concentration for Retrieval Fluids for Single-Shell Tanks within TX Tank Farm

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO₂ kg	NO₃ kg	U kg
TX-101	3.92E-01	3.10E-02	6.03E-04	8.36E-01	3.59E+01	1.79E+01	1.06E+03	4.78E+03	4.43E+00
TX-102	2.38E-01	3.60E-02	1.62E-03	8.48E-01	2.02E+01	4.48E+01	6.16E+02	6.12E+03	1.46E+00
TX-103	2.39E-01	3.60E-02	1.55E-03	7.62E-01	2.00E+01	4.86E+01	5.79E+02	6.21E+03	3.00E-01
TX-104	2.67E-01	2.77E-02	1.49E-03	9.22E-01	4.59E+01	4.27E+01	8.01E+02	7.46E+03	5.37E-01
Sum of rows TX101-TX104	1.14E+00	1.31E-01	5.26E-03	3.37E+00	1.22E+02	1.54E+02	3.06E+03	2.46E+04	6.73E+00
TX-105	2.58E-01	3.84E-02	1.43E-03	5.70E-01	1.97E+01	4.83E+01	5.08E+02	6.41E+03	1.98E+00
TX-106	2.35E-01	3.70E-02	1.49E-03	1.40E+00	1.92E+01	4.37E+01	5.61E+02	5.94E+03	4.63E-03
TX-107	3.42E-01	3.33E-02	1.69E-03	8.56E-01	2.72E+01	3.17E+01	6.18E+02	6.14E+03	2.47E-01
TX-108	2.47E-01	3.69E-02	1.48E-03	6.64E-01	1.96E+01	4.65E+01	5.46E+02	6.28E+03	2.45E+00
Sum of rows TX105-TX108	1.08E+00	1.46E-01	6.09E-03	3.49E+00	8.56E+01	1.70E+02	2.23E+03	2.48E+04	4.69E+00
TX-109	1.34E-02	8.15E-03	1.82E-05	8.61E-01	5.01E+00	3.62E+02	3.44E+02	4.47E+03	0.00E+00
TX-110	2.54E-01	3.79E-02	1.35E-03	5.14E-01	1.95E+01	6.14E+01	4.86E+02	6.44E+03	3.64E-01
TX-111	2.51E-01	3.75E-02	1.31E-03	4.83E-01	1.91E+01	6.81E+01	4.73E+02	6.44E+03	3.88E-01
TX-112	2.52E-01	3.73E-02	1.37E-03	5.28E-01	1.93E+01	5.37E+01	4.84E+02	6.44E+03	3.19E-01
Sum of rows TX109-TX112	7.70E-01	1.21E-01	4.05E-03	2.39E+00	6.29E+01	5.45E+02	1.79E+03	2.38E+04	1.07E+00
TX-113	2.47E-01	3.69E-02	1.26E-03	4.42E-01	4.85E+00	8.44E+01	6.27E+01	6.71E+03	1.40E-01
TX-114	2.39E-01	3.51E-02	1.23E-03	4.30E-01	1.77E+01	6.39E+01	4.29E+02	6.43E+03	3.31E-01
TX-115	2.58E-01	3.83E-02	1.43E-03	5.71E-01	2.00E+01	4.85E+01	5.11E+02	6.45E+03	3.76E-01
Sum of rows TX113-TX115	7.43E-01	1.10E-01	3.92E-03	1.44E+00	4.25E+01	1.97E+02	1.00E+03	1.96E+04	8.46E-01
TX-116	1.15E-01	1.64E-02	7.63E-04	2.61E-01	2.07E+00	5.33E+01	2.01E+02	7.26E+03	2.92E-01
TX-117	1.64E-01	2.35E-02	7.91E-04	2.58E-01	1.18E+01	1.01E+02	2.82E+02	6.12E+03	3.21E-01
TX-118	1.85E-01	2.81E-02	1.34E-03	7.36E-01	1.60E+01	2.96E+02	4.22E+02	1.44E+03	3.30E-01
Sum of rows TX116-TX118	4.64E-01	6.80E-02	2.89E-03	1.25E+00	2.99E+01	4.51E+02	9.05E+02	1.48E+04	9.42E-01

[a] Kirkbride et al. (2005)

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**Table C-20. Summary of 8,000-Gallon Retrieval Leak Inventory Estimates Based
on HTWOS [a] Model Estimated Concentration for Retrieval Fluids for
Single-Shell Tanks within TY Tank Farm**

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO$_2$ kg	NO$_3$ kg	U kg
TY-101	1.26E-03	7.06E-04	4.03E-06	1.25E-02	1.79E+01	9.90E+00	3.36E+01	9.24E+01	3.34E-02
TY-102	1.32E-01	8.46E-03	5.17E-04	3.72E-01	3.38E+00	9.78E+01	3.96E+02	8.00E+03	1.62E-01
Sum of rows TY101-TY102	1.33E-01	9.16E-03	5.21E-04	3.85E-01	2.13E+01	1.08E+02	4.29E+02	8.09E+03	1.95E-01
TY-103	2.68E-02	3.74E-03	2.10E-04	1.86E-01	3.73E+00	1.12E+01	1.06E+02	1.19E+03	3.65E-01
TY-104	3.35E-03	1.18E-02	8.62E-06	1.53E-01	6.54E+00	2.83E+01	7.99E+01	3.13E+02	2.53E-01
Sum of rows TY103-TY104	3.01E-02	1.55E-02	2.19E-04	3.39E-01	1.03E+01	3.94E+01	1.86E+02	1.50E+03	6.17E-01
TY-105	1.69E-02	7.74E-04	3.47E-04	3.32E-01	2.42E-01	6.71E+00	1.58E+02	1.80E+03	1.42E-01
TY-106	1.15E-03	5.18E-05	7.57E-07	3.02E-02	1.35E-03	1.88E-01	2.29E+01	4.46E+02	2.80E-02
Sum of rows TY105-TY106	1.81E-02	8.26E-04	3.48E-04	3.62E-01	2.43E-01	6.90E+00	1.81E+02	2.24E+03	1.70E-01

[a] Kirkbride et al. (2005)

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1 **C5.0 WASTE MANAGEMENT AREA U INVENTORY**

2 This section provides the inventory of the major risk-driving radionuclides and hazardous
3 chemicals found within WMA U.

4 **C5.1 SINGLE-SHELL TANK RESIDUAL WASTE INVENTORY**

**Table C-21. Summary of Final Tank Residual Inventory Estimates Based on
HTWOS[a] Model Assuming 10.2 kL (360 ft[3]) Remaining in the Tank
in Waste Management Area U**

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO$_2$ kg	NO$_3$ kg	U kg
U-101	2.52E-02	3.48E-03	4.58E-05	2.86E-02	6.69E+01	4.19E-01	6.56E+01	2.32E+02	1.69E+02
U-102	3.53E-02	1.58E-02	2.40E-04	3.66E+00	1.17E+02	8.58E-01	1.53E+02	8.00E+02	8.14E+01
U-103	2.44E-02	6.72E-03	2.29E-04	1.31E+00	4.10E+02	2.32E+01	1.36E+02	5.93E+02	4.65E+01
Sum of rows U101-U103	8.49E-02	2.60E-02	5.15E-04	5.01E+00	5.94E+02	2.45E+01	3.54E+02	1.63E+03	2.97E+02
U-104	1.79E-02	1.26E-03	3.26E-05	2.04E-02	5.01E+01	1.07E-01	3.42E+01	1.28E+02	4.82E+01
U-105	4.66E-02	8.24E-03	2.02E-03	9.75E-01	1.34E+02	4.70E+00	1.30E+02	8.02E+02	3.42E+02
U-106	2.16E-02	6.14E-03	2.78E-04	2.93E-01	3.09E+02	1.48E+00	1.47E+02	5.67E+02	8.96E+01
Sum of rows U104-U106	8.61E-02	1.56E-02	2.33E-03	1.29E+00	4.92E+02	6.29E+00	3.11E+02	1.50E+03	4.80E+02
U-107	4.10E-02	2.57E-02	3.14E-04	6.62E+00	2.45E+02	1.28E+00	1.08E+02	9.47E+02	3.16E+01
U-108	5.15E-02	1.55E-02	2.57E-04	3.07E+00	1.72E+02	2.27E+00	1.42E+02	7.39E+02	8.11E+01
U-109	4.01E-02	2.48E-02	2.68E-04	6.73E+00	2.89E+02	5.08E+00	1.34E+02	7.27E+02	1.65E+01
U-201	8.26E-02	4.22E-04	2.78E-07	3.76E-03	3.55E-01	8.77E+00	3.58E+00	2.38E+01	1.88E-02
U-202	7.58E-02	3.06E-04	2.55E-07	3.41E-03	2.90E-01	9.59E+00	3.31E+00	2.01E+01	1.84E-02
Sum of rows U107-U109 and U201-U202	2.91E-01	6.67E-02	8.39E-04	1.64E+01	7.07E+02	2.70E+01	3.91E+02	2.46E+03	1.29E+02
U-110	5.36E-04	1.98E-04	7.19E-06	4.53E-02	1.79E+00	3.06E+00	6.83E+00	2.37E+01	1.88E+02
U-111	2.83E-02	2.31E-02	2.74E-04	6.23E+00	4.04E+02	3.27E+00	1.47E+02	7.17E+02	8.05E+01
U-112	1.66E-02	2.81E-04	5.57E-06	7.42E-03	6.43E+00	5.01E+00	2.05E+00	3.92E+01	5.72E+01
U-203	6.62E-02	2.67E-04	2.23E-07	3.34E-03	3.15E-01	1.82E+01	3.30E+00	2.19E+01	9.34E-03
U-204	7.54E-02	3.05E-04	2.54E-07	1.99E-03	1.75E-01	6.38E-02	2.34E+00	1.62E+01	1.31E+00
Sum of rows U110-U112 and U203-U204	1.87E-01	2.42E-02	2.88E-04	6.28E+00	4.12E+02	2.96E+01	1.61E+02	8.18E+02	3.27E+02

[a] Kirkbride et al. (2005)

5

C5.2 PAST TANK LEAKS AND UNPLANNED RELEASES INVENTORY

Table C-22. Summary of Past Leak Inventory Estimates within Waste Management Area U Based on Volume Estimates [a] and Soil Inventory Model [b]

Analyte→ Tank↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO₂ kg	NO₃ kg	NH₃ kg	Butanol kg	U kg
UPR-200-W-132	4.90E-02	8.16E-04	2.49E-05	2.18E-02	3.51E-01	0.00E+00	2.00E+00	2.15E+01	4.41E-03	0.00E+00	1.80E+00
UPR-200-W-24	1.59E-03	4.25E-05	4.18E-07	9.44E-04	2.51E-02	0.00E+00	1.05E-01	1.58E+00	1.68E-04	0.00E+00	1.31E-01
Sum of rows UR vault row	5.06E-02	8.58E-04	2.53E-05	2.27E-02	3.76E-01	0.00E+00	2.11E+00	2.31E+01	4.58E-03	0.00E+00	1.93E+00
U-101	4.79E-01	2.62E-02	7.36E-04	4.60E-01	4.99E+01	2.15E-04	6.14E+02	2.42E+03	2.82E+00	0.00E+00	5.38E-01
Sum of rows U101-U103	4.79E-01	2.62E-02	7.36E-04	4.60E-01	4.99E+01	2.15E-04	6.14E+02	2.42E+03	2.82E+00	0.00E+00	5.38E-01
U-104	4.72E+00	7.99E-02	2.37E-03	2.12E+00	3.49E+01	2.69E-03	1.96E+02	2.14E+03	4.27E-01	0.00E+00	1.79E+02
Sum of rows U104-U106	4.72E+00	7.99E-02	2.37E-03	2.12E+00	3.49E+01	2.69E-03	1.96E+02	2.14E+03	4.27E-01	0.00E+00	1.79E+02
U-110	3.98E-01	1.62E-02	4.15E-04	3.74E-01	8.33E+00	6.43E+00	3.47E+02	6.92E+02	5.23E+00	1.78E+00	1.53E-01
U-112	8.42E-01	3.74E-02	9.82E-04	6.18E-01	6.77E+01	1.01E+01	9.05E+02	3.59E+03	9.12E+00	0.00E+00	1.57E+00
Sum of rows U110-U112 and U203-U204	1.24E+00	5.35E-02	1.40E-03	9.92E-01	7.61E+01	1.66E+01	1.25E+03	4.28E+03	1.44E+01	1.78E+00	1.72E+00

[a] Field and Jones (2005)

[b] Corbin et al. (2005)

C5.3 ANCILLARY EQUIPMENT (PIPELINES AND MUST) INVENTORY

Table C-23. Summary of Final Pipeline Inventory Estimates for Waste Management Area U [a]

Analyte→ Tank↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO₂ kg	NO₃ kg	U kg
Pipeline	1.23E-02	8.94E-03	1.50E-04	1.02E-01	2.42E+00	3.73E-02	5.45E+01	1.19E+02	1.41E-02

[a] Lambert (2005)

Table C-24. Summary of Extrapolated MUST Inventory for Waste Management Area U

Analyte→ Tank↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO₂ kg	NO₃ kg	U kg
244-UR vault	6.95E-03	1.42E-03	4.25E-05	3.10E-01	2.36E+01	9.35E-01	1.30E+01	6.84E+01	1.32E+01
241-U-DCR placed in tank row U110–U112	1.68E-03	3.43E-04	1.03E-05	7.51E-02	5.71E+00	2.26E-01	3.15E+00	1.66E+01	3.19E+00

1 **C5.4   POTENTIAL RETRIEVAL LEAKS**

Table C-25. Summary of 8,000-Gallon Retrieval Leak Inventory Estimates Based on HTWOS [a] Model Estimated Concentration for Retrieval Fluids for Single-Shell Tanks within Waste Management Area U

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO₂ kg	NO₃ kg	U kg
U-101	1.84E-01	1.10E-02	3.34E-04	2.09E-01	1.70E+01	3.05E+00	4.78E+02	1.69E+03	9.63E-12
U-102	2.54E-01	4.35E-02	1.73E-03	8.19E-01	2.33E+01	6.18E+00	1.02E+03	5.45E+03	4.29E-01
U-103	1.75E-01	4.82E-02	1.64E-03	1.59E+00	7.88E+00	9.70E+00	9.19E+02	3.73E+03	4.18E+00
Sum of rows U101-U103	6.13E-01	1.03E-01	3.70E-03	2.62E+00	4.81E+01	1.89E+01	2.42E+03	1.09E+04	4.61E+00
U-104	1.27E-01	8.20E-03	2.31E-04	1.45E-01	2.54E+00	7.57E-01	2.42E+02	9.10E+02	2.91E-04
U-105	3.34E-01	5.76E-02	1.52E-03	1.87E+00	5.70E+00	2.32E+01	8.65E+02	5.49E+03	1.52E+00
U-106	1.55E-01	4.39E-02	1.99E-03	2.10E+00	5.32E+00	1.06E+01	1.05E+03	4.05E+03	9.03E-03
Sum of rows U104-U106	6.16E-01	1.10E-01	3.74E-03	4.12E+00	1.36E+01	3.46E+01	2.15E+03	1.05E+04	1.52E+00
U-107	2.94E-01	6.18E-02	2.25E-03	1.08E+00	7.82E+00	9.16E+00	7.11E+02	6.34E+03	1.37E-01
U-108	3.67E-01	5.43E-02	1.86E-03	8.24E-01	7.39E+00	1.62E+01	9.80E+02	5.10E+03	6.62E-01
U-109	2.88E-01	5.33E-02	1.92E-03	8.98E-01	1.20E+01	3.64E+01	8.65E+02	4.82E+03	6.25E-02
Sum of rows U107-U109	9.49E-01	1.69E-01	6.03E-03	2.81E+00	2.72E+01	6.18E+01	2.56E+03	1.63E+04	8.61E-01
U-110	3.76E-03	1.39E-03	5.05E-05	3.04E-02	1.70E+00	2.15E+01	4.79E+01	1.66E+02	4.59E-04
U-111	2.04E-01	5.18E-02	1.97E-03	9.82E-01	9.80E+00	2.35E+01	9.59E+02	4.79E+03	3.12E-01
U-112	1.17E-01	1.98E-03	3.93E-05	5.23E-02	4.47E-01	3.53E+01	1.45E+01	2.76E+02	2.04E-08
Sum of rows U110-U112	3.24E-01	5.52E-02	2.06E-03	1.07E+00	1.19E+01	8.03E+01	1.02E+03	5.23E+03	3.13E-01

[a] Kirkbride et al. (2005)

2

1 **C6.0 WASTE MANAGEMENT AREA C INVENTORY**

2 This section provides the inventory of the major risk-driving radionuclides and hazardous
3 chemicals found within WMA C.

4 **C6.1 SINGLE-SHELL TANK RESIDUAL WASTE INVENTORY**

Table C-26. Summary of Final Tank Residual Inventory Estimates Based on HTWOS [a] Model Assuming 10.2 kL (360 ft[3]) Remaining in the Tank in Waste Management Area C

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO₂ kg	NO₃ kg	U kg
C-101	2.93E-01	3.65E-03	1.68E-03	2.79E-03	4.18E+00	3.50E+00	3.84E+01	3.37E+02	1.93E+02
C-104	1.79E-01	7.22E-03	4.23E-03	2.35E-01	9.75E+00	1.31E+02	1.38E+02	7.40E+01	2.45E+02
C-107	1.19E-02	3.46E-03	2.72E-03	2.13E-01	6.25E+00	2.44E+01	1.34E+02	3.34E+02	6.80E+01
C-110	1.68E-03	1.18E-03	1.18E-06	1.67E-01	7.83E+00	1.40E+02	3.39E+01	5.62E+02	6.09E+01
Sum of rows C101-C110	4.86E-01	1.55E-02	8.64E-03	6.18E-01	2.80E+01	2.99E+02	3.44E+02	1.31E+03	5.67E+02
C-102	1.10E-01	4.72E-03	1.58E-03	3.20E-03	5.83E+00	1.26E+01	5.78E+01	1.99E+02	1.48E+02
C-105	6.73E-01	3.28E-03	5.99E-04	5.57E-01	2.56E+00	5.30E+00	4.34E+01	5.24E+01	7.39E+01
C-108	8.15E-03	5.90E-04	7.03E-06	4.05E-02	5.92E+00	1.13E+01	6.43E+01	1.16E+02	6.67E+00
C-111	9.02E-02	1.31E-03	2.00E-04	1.58E-02	2.19E+00	5.74E+00	6.58E+01	1.27E+02	1.42E+02
Sum of rows C102-C111	8.81E-01	9.90E-03	2.39E-03	6.16E-01	1.65E+01	3.50E+01	2.31E+02	4.94E+02	3.70E+02
C-103	1.85E-01	3.94E-03	4.63E-04	7.17E-02	6.02E+00	3.61E+00	5.94E+01	5.81E+00	4.55E+01
C-106	1.02E-02	8.23E-03	6.31E-04	1.65E-01	3.78E+00	5.43E-01	4.14E+01	4.60E+01	2.70E+00
C-109	9.24E-03	1.30E-04	4.28E-04	4.96E-01	1.68E+00	1.00E+01	1.03E+02	1.61E+02	1.46E+02
C-112	3.64E-02	2.29E-02	3.41E-04	7.23E-01	1.84E+00	5.08E+00	1.74E+02	2.45E+02	6.71E+02
Sum of rows C103-C112	2.41E-01	3.52E-02	1.86E-03	1.46E+00	1.33E+01	1.93E+01	3.78E+02	4.57E+02	8.65E+02
C-201	5.81E-04	1.01E-03	6.88E-07	1.21E-02	8.66E+00	2.78E+00	3.04E+01	8.60E+01	1.69E+02
C-202	5.63E-04	2.89E-04	6.66E-07	1.13E-02	6.32E+00	2.28E+00	2.54E+01	7.77E+01	1.47E+02
C-203	5.58E-04	2.87E-04	6.61E-07	1.88E-03	1.69E+01	6.80E+00	7.96E+00	1.60E+02	2.19E+02
C-204	4.19E-04	2.31E-04	5.31E-07	6.02E-03	9.86E+00	1.36E-01	2.17E+01	3.13E+01	8.82E+01
Sum of rows C201-C204	2.12E-03	1.82E-03	2.55E-06	3.13E-02	4.18E+01	1.20E+01	8.54E+01	3.55E+02	6.23E+02

[a] Kirkbride et al. (2005)

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1   ## C6.2   PAST TANK LEAKS AND UNPLANNED RELEASES INVENTORY

Table C-27. Summary of Past Leak Inventory Estimates within WMA C Based on Volume Estimates [a] and Soil Inventory Model [b]

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO$_2$ kg	NO$_3$ kg	NH$_3$ kg	Butanol kg	U kg
UPR-200-E-81	2.29E+01	1.46E-01	2.38E-02	2.74E-02	2.17E+01	0.00E+00	1.78E+03	5.81E+03	4.29E-02	0.00E+00	4.22E+00
UPR-200-E-82	1.94E-01	3.73E-02	8.39E-04	1.42E+00	1.62E+01	3.60E-01	3.93E+02	7.64E+02	4.14E+00	0.00E+00	3.04E-01
UPR-200-E-86	7.21E-01	1.31E-01	2.61E-03	4.92E+00	6.04E+01	8.43E-01	1.49E+03	1.27E+03	2.04E+01	0.00E+00	2.11E+00
Sum of rows CR vault row	2.38E+01	3.14E-01	2.73E-02	6.36E+00	9.83E+01	1.20E+00	3.67E+03	7.84E+03	2.46E+01	0.00E+00	6.63E+00
C-101	4.09E-02	7.68E-03	1.21E-04	2.25E-01	1.51E+00	1.82E-02	4.07E+01	5.03E+01	4.39E-01	1.11E+00	1.16E-01
C-110	4.28E-02	1.64E-03	3.11E-05	2.43E-02	1.47E+00	4.13E+00	5.08E+01	1.30E+03	2.23E+00	4.41E-03	2.35E-01
UPR-200-E-107	1.16E-04	4.49E-06	9.06E-08	7.07E-05	3.14E-03	0.00E+00	1.39E-01	3.44E+00	1.29E-04	1.29E-05	5.87E-04
Sum of rows C101-C110	8.38E-02	9.33E-03	1.52E-04	2.49E-01	2.99E+00	4.15E+00	9.17E+01	1.35E+03	2.67E+00	1.11E+00	3.51E-01
C-105	4.19E-02	5.00E-03	8.85E-05	2.26E-01	1.42E+00	8.33E-02	3.77E+01	1.14E+02	5.03E-01	6.47E-01	9.86E-02
C-111	1.58E+00	1.63E-02	2.64E-03	5.38E-02	5.27E+00	2.68E+00	4.66E+02	1.07E+03	1.19E+00	8.02E-03	6.01E-01
Sum of rows C102-C111	1.62E+00	2.13E-02	2.73E-03	2.80E-01	6.69E+00	2.77E+00	5.03E+02	1.19E+03	1.70E+00	6.55E-01	7.00E-01
C-201	6.41E-04	2.78E-04	6.42E-07	1.07E-02	7.80E-01	1.67E-05	2.63E+01	9.69E+01	1.20E-01	0.00E+00	6.08E-02
C-202	5.24E-04	2.27E-04	5.25E-07	8.77E-03	6.37E-01	1.39E-05	2.15E+01	7.92E+01	9.83E-02	0.00E+00	4.96E-02
C-203	4.96E-04	2.15E-04	4.96E-07	8.29E-03	6.03E-01	0.00E+00	2.03E+01	7.49E+01	9.30E-02	0.00E+00	4.69E-02
C-204	4.21E-04	1.82E-04	4.21E-07	7.04E-03	5.12E-01	5.31E-06	1.72E+01	6.36E+01	7.90E-02	0.00E+00	3.99E-02
Sum of rows C201-C204	2.08E-03	9.01E-04	2.08E-06	3.48E-02	2.53E+00	3.59E-05	8.53E+01	3.15E+02	3.91E-01	0.00E+00	1.97E-01

[a] Field and Jones (2005)

[b] Corbin et al. (2005)

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3   ## C6.3   ANCILLARY EQUIPMENT (PIPELINES AND MUST) INVENTORY

Table C-28. Summary of Final Pipeline Inventory Estimates for Waste Management Area C [a]

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO$_2$ kg	NO$_3$ kg	U kg
Pipeline	8.86E-04	1.78E-04	3.03E-07	3.58E-05	1.05E-02	1.10E-01	7.13E-02	3.13E+00	4.36E-04

[a] Lambert (2005)

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Table C-29. Summary of Extrapolated MUST Inventory for Waste Management Area C

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO₂ kg	NO₃ kg	U kg
244-CR vault	2.72E-02	1.06E-03	2.18E-04	4.60E-02	1.68E+00	6.17E+00	1.76E+01	4.42E+01	4.10E+01
241-C-301 placed in tank row C103-C112	7.39E-03	2.87E-04	5.92E-05	1.25E-02	4.57E-01	1.68E+00	4.77E+00	1.20E+01	1.11E+01

C6.4 POTENTIAL RETRIEVAL LEAKS

Table C-30. Summary of 8,000-Gallon Retrieval Leak Inventory Estimates Based on HTWOS [a] Model Estimated Concentration for Retrieval Fluids for Single-Shell Tanks within Waste Management Area C

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO₂ kg	NO₃ kg	U kg
C-101	2.08E+00	2.30E-02	2.12E-03	1.50E-02	3.60E-01	2.48E+01	2.73E+02	2.40E+03	4.23E+00
C-104	1.27E+00	4.81E-02	8.57E-03	1.42E+00	3.53E+00	9.30E+02	9.82E+02	5.26E+02	1.15E+00
C-107	8.44E-02	7.22E-03	1.92E-02	6.15E-01	9.21E+00	1.69E+02	9.45E+02	4.00E+02	5.94E+01
C-110	1.19E-02	6.79E-03	8.34E-06	6.11E-01	4.03E+00	5.48E+01	1.26E+02	1.85E+03	1.78E+00
Sum of rows C101-C110	3.45E+00	8.51E-02	3.00E-02	2.67E+00	1.71E+01	1.18E+03	2.33E+03	5.17E+03	6.65E+01
C-102	7.81E-01	3.00E-02	2.12E-03	1.75E-02	1.11E+00	8.96E+01	4.10E+02	1.41E+03	3.49E+00
C-105	4.74E+00	2.22E-02	4.22E-03	2.39E+00	1.41E+01	3.73E+01	3.05E+02	3.69E+02	2.40E+01
C-108	5.77E-02	4.18E-03	4.98E-05	2.86E-01	6.00E+00	8.02E+01	4.55E+02	8.20E+02	7.95E-01
C-111	6.36E-01	9.25E-03	1.41E-03	1.12E-01	9.68E-01	4.04E+01	4.64E+02	8.96E+02	0.00E+00
Sum of rows C102-C111	6.21E+00	6.56E-02	7.80E-03	2.81E+00	2.22E+01	2.48E+02	1.63E+03	3.49E+03	2.82E+01
C-103	1.30E+00	2.66E-02	3.25E-03	5.04E-01	5.91E-01	2.54E+01	4.18E+02	4.08E+01	7.70E+01
C-106	2.60E-04	2.05E-04	1.14E-05	6.72E-04	1.92E-03	1.19E-02	1.04E+00	1.13E+00	1.54E-03
C-109	6.53E-02	8.58E-04	1.09E-03	1.37E+00	5.09E+00	2.74E+01	6.91E+02	1.01E+03	1.72E+00
C-112	2.58E-01	1.50E-01	8.29E-04	1.92E+00	4.00E+00	1.33E+01	1.16E+03	1.53E+03	4.12E+01
Sum of rows C103-C112	1.62E+00	1.78E-01	5.18E-03	3.79E+00	9.69E+00	6.61E+01	2.27E+03	2.58E+03	1.20E+02

[a] Kirkbride et al. (2005)

1 **C7.0   WASTE MANAGEMENT AREA B-BX-BY INVENTORY**

2 This section provides the inventory of the major risk-driving radionuclides and hazardous
3 chemicals found within WMA B-BX-BY.

4 **C7.1   SINGLE-SHELL TANK RESIDUAL WASTE INVENTORY**

**Table C-31. Summary of Final Tank Residual Inventory Estimates Based on
HTWOS [a] Model Assuming 10.2 kL (360 ft^3) Remaining in the Tank
in B and BX Tank Farms (2 pages)**

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO$_2$ kg	NO$_3$ kg	U kg
B-101	3.34E-03	2.37E-04	3.07E-06	1.22E-02	1.86E+01	6.81E-01	9.44E+01	3.42E+02	3.94E+02
B-104	1.29E-03	3.85E-04	1.57E-06	3.52E-02	3.24E+01	9.41E+01	9.63E+00	1.04E+03	1.29E+02
B-107	7.62E-03	1.70E-03	8.43E-05	1.17E-01	6.48E+00	4.33E+02	2.63E+01	1.25E+03	5.51E+01
B-110	8.00E-03	1.01E-02	5.28E-06	1.62E-01	2.96E+01	5.05E+01	1.04E+02	1.97E+03	8.06E+00
Sum of rows B101-B110	2.02E-02	1.24E-02	9.42E-05	3.26E-01	8.71E+01	5.78E+02	2.34E+02	4.61E+03	5.87E+02
B-102	2.94E-03	2.04E-04	3.27E-06	2.35E-03	2.28E+01	3.58E+02	1.48E+01	2.44E+02	8.06E+01
B-105	2.11E-03	1.64E-04	2.35E-06	3.84E-03	3.56E+01	5.09E+02	2.00E+01	2.57E+02	1.20E+02
B-108	2.45E-03	4.23E-04	5.90E-05	8.63E-04	2.06E+00	2.98E+02	1.74E+01	1.00E+02	5.39E+01
B-111	1.08E-02	1.22E-02	2.77E-06	1.08E+00	3.49E+01	4.17E+01	4.24E+02	8.24E+02	2.22E+00
BX-101	1.42E-03	3.97E-04	3.14E-04	1.24E-03	4.76E-01	2.23E+00	6.20E+00	5.83E+01	6.65E+01
BX-104	1.43E-02	1.16E-04	2.19E-05	1.09E-01	2.85E+01	4.01E-01	6.04E+01	1.22E+02	5.49E+02
BX-107	3.37E-04	4.24E-04	6.09E-07	1.02E-01	1.79E+01	1.13E+02	2.34E+01	3.95E+02	3.92E+01
BX-110	4.02E-02	9.50E-03	8.87E-04	1.35E+00	1.65E+02	1.57E+01	3.66E+01	8.61E+02	3.98E+01
Sum of rows B102-B111 and BX101-BX110	7.46E-02	2.35E-02	1.29E-03	2.65E+00	3.07E+02	1.34E+03	6.03E+02	2.86E+03	9.51E+02
B-103	2.28E-03	1.57E-04	2.48E-06	1.82E-03	2.20E+01	3.46E+02	1.30E+01	2.02E+02	2.70E+02
B-106	1.88E-03	2.53E-04	1.87E-06	2.01E-02	1.49E+01	8.24E+01	1.65E+01	4.11E+02	4.45E+02
B-109	2.17E-03	3.96E-04	5.70E-05	6.79E-04	1.59E+01	2.07E+02	6.42E+00	1.18E+02	2.23E+02
B-112	2.53E-02	4.10E-03	3.96E-04	1.23E-01	4.10E+01	9.00E+01	1.22E+02	6.85E+02	8.75E+00
BX-102	6.20E-04	1.98E-04	1.62E-04	1.71E-04	1.23E-01	7.74E-01	1.94E+00	7.55E+00	1.62E+01
BX-105	7.70E-02	1.20E-03	1.61E-05	1.94E-01	4.27E+01	9.83E+00	2.31E+01	6.15E+01	5.79E+02
BX-108	1.70E-03	2.68E-04	1.66E-06	3.51E-02	1.04E+01	7.90E+01	4.70E+01	7.59E+02	2.61E+02
BX-111	4.85E-02	6.62E-02	1.07E-02	3.04E+00	7.38E+01	8.16E+01	4.69E+01	1.46E+03	1.88E+01
Sum of rows B103-B112 and BX102-BX111	1.59E-01	7.28E-02	1.13E-02	3.41E+00	2.21E+02	8.97E+02	2.77E+02	3.71E+03	1.82E+03
B-201	1.66E-02	1.51E-07	6.07E-12	1.37E-08	3.87E+00	6.65E+00	9.77E-01	5.60E+01	1.73E-01
B-202	4.28E-09	2.09E-07	0.00E+00	5.54E-03	2.88E+00	6.65E+00	1.17E+00	6.55E+01	3.45E-01
B-203	4.21E-09	2.06E-07	0.00E+00	1.89E-08	3.21E+00	7.87E+00	7.34E-01	6.46E+01	8.63E-03
B-204	3.93E-09	1.92E-07	5.83E-19	1.77E-08	3.41E+00	6.74E+00	5.98E-01	5.13E+01	6.30E-06

Table C-31. Summary of Final Tank Residual Inventory Estimates Based on HTWOS [a] Model Assuming 10.2 kL (360 ft³) Remaining in the Tank in B and BX Tank Farms (2 pages)

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO₂ kg	NO₃ kg	U kg
BX-103	1.77E-02	5.87E-04	3.80E-04	1.87E-03	8.32E-01	2.90E+00	9.81E+00	9.27E+01	8.88E+01
BX-106	3.08E-02	4.68E-04	7.02E-05	1.39E-01	5.68E+01	2.01E+00	1.15E+01	3.53E+01	5.96E+01
BX-109	3.41E-03	3.04E-04	3.14E-06	3.11E-02	5.27E+00	6.65E-01	4.74E+01	5.06E+02	6.90E+02
BX-112	4.56E-04	3.22E-04	5.39E-07	6.07E-02	2.90E+01	3.02E+02	2.68E+02	7.41E+02	2.60E+01
Sum of rows B201-B204 and BX103-BX112	6.91E-02	1.68E-03	4.54E-04	2.38E-01	1.05E+02	3.35E+02	3.40E+02	1.61E+03	8.65E+02

[a] Kirkbride et al. (2005)

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Table C-32. Summary of Final Tank Residual Inventory Estimates Based on the HTWOS [a] Model Assuming 10.2 kL (360 ft³) Remaining in the Tank in BY Tank Farm

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO₂ kg	NO₃ kg	U kg
BY-101	5.19E-02	5.67E-02	9.37E-03	2.43E+00	4.27E+01	8.33E+01	2.21E+01	1.17E+03	6.29E+01
BY-104	5.21E-02	1.06E-01	1.72E-02	4.70E+00	1.90E+02	1.44E+02	1.39E+02	1.09E+03	4.41E+02
BY-107	4.59E-02	1.29E-02	1.13E-03	1.89E+00	1.76E+01	8.73E+00	1.05E+02	4.44E+02	1.55E+02
BY-110	4.87E-02	7.53E-02	1.23E-02	3.29E+00	6.59E+01	1.22E+02	1.21E+02	4.75E+02	3.21E+02
Sum of rows BY101-BY110	1.99E-01	2.51E-01	3.99E-02	1.23E+01	3.17E+02	3.59E+02	3.86E+02	3.18E+03	9.81E+02
BY-102	4.78E-02	8.94E-03	1.03E-02	9.39E-01	4.02E+01	3.14E+02	5.37E+01	3.72E+02	2.42E+01
BY-105	5.94E-02	5.72E-02	9.52E-03	3.95E-01	1.32E+01	3.10E+01	3.65E+01	1.56E+03	1.16E+02
BY-108	5.84E-02	1.24E-02	1.33E-03	1.33E+00	8.56E+00	2.19E+01	8.58E+01	5.04E+02	3.87E+02
BY-111	4.78E-02	1.09E-02	1.24E-03	1.27E+00	3.97E+01	1.95E+01	3.85E+01	3.18E+02	1.42E+01
Sum of rows BY102-BY111	2.13E-01	8.94E-02	2.24E-02	3.94E+00	1.02E+02	3.86E+02	2.14E+02	2.75E+03	5.41E+02
BY-103	5.30E-02	5.51E-02	9.08E-03	2.34E+00	5.85E+01	1.72E+02	2.79E+01	7.46E+02	2.35E+01
BY-106	5.13E-02	5.11E-02	1.58E-03	1.19E+01	1.63E+02	1.49E+01	9.20E+01	9.67E+02	6.61E+02
BY-109	5.27E-02	3.63E-02	5.89E-03	1.43E+00	3.24E+01	4.15E+02	9.19E+01	2.68E+02	1.30E+01
BY-112	3.79E-02	6.94E-03	7.66E-03	6.90E-01	4.04E+02	1.03E+02	5.89E+01	2.02E+02	9.68E+01
Sum of rows BY103-BY112	1.95E-01	1.49E-01	2.42E-02	1.64E+01	6.58E+02	7.04E+02	2.71E+02	2.18E+03	7.95E+02

[a] Kirkbride et al. (2005)

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1 **C7.2 PAST TANK LEAKS AND UNPLANNED RELEASES INVENTORY**

Table C-33. Summary of Past Leak Inventory Estimates within B and BX Tank Farms Based on Volume Estimates [a] and Soil Inventory Model [b]

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO_2 kg	NO_3 kg	NH_3 kg	Butanol kg	U kg
B-107	6.94E+00	5.72E-02	8.24E-03	2.83E-01	9.29E+00	8.57E+00	8.07E+02	2.35E+03	4.37E+00	1.42E+00	1.47E+00
B-110	1.20E-01	1.68E-02	4.09E-04	5.42E-01	5.56E+00	2.15E+00	1.92E+02	3.40E+02	1.85E+00	1.69E+00	1.10E-01
UPR-200-E-6	2.35E-02	3.12E-04	6.81E-06	7.78E-04	1.38E+00	1.44E+01	8.93E+00	4.13E+02	5.56E+00	0.00E+00	1.19E-01
UPR-200-E-73	4.50E-03	7.45E-05	2.28E-06	2.01E-03	3.20E-02	0.00E+00	1.83E-01	1.97E-00	4.05E-04	0.00E+00	1.65E-01
UPR-200-E-75	2.35E-02	3.11E-04	6.79E-06	7.76E-04	1.38E+00	1.44E+01	8.92E+00	4.13E+02	5.55E+00	0.00E+00	1.20E-01
UPR-200-E-109	3.48E-03	1.35E-04	2.72E-06	2.12E-03	9.42E-02	0.00E+00	4.16E+00	1.03E+02	3.86E-03	3.87E-04	1.76E-02
UPR-200-E-74	4.57E-08	8.84E-10	1.88E-11	1.88E-08	1.60E-02	0.00E+00	4.25E-02	3.74E-01	0.00E+00	0.00E+00	1.91E-06
UPR-200-E-38	4.02E-01	7.75E-02	1.74E-03	2.94E+00	3.36E+01	7.48E-01	8.16E+02	1.59E+03	8.60E+00	0.00E+00	6.33E-01
Sum of rows B101-B110	7.52E+00	1.52E-01	1.04E-02	3.77E+00	5.14E+01	4.03E+01	1.84E+03	5.21E+03	2.59E+01	3.11E+00	2.64E+00
BX-101	4.81E-02	5.22E-03	9.06E-05	2.47E-01	1.50E+00	3.66E-02	1.09E+02	7.24E+02	5.28E-01	2.51E-02	4.54E-01
UPR-200-E-108	1.92E-02	3.19E-04	9.76E-06	8.54E-03	1.38E-01	0.00E+00	7.85E-01	8.42E+00	1.73E-03	0.00E+00	7.09E-01
Sum of rows B102-B111 and BX101-BX110	6.73E-02	5.53E-03	1.00E-04	2.56E-01	1.64E+00	3.66E-02	1.10E+02	7.32E+02	5.29E-01	2.51E-02	1.16E+00
B-112	1.81E-01	2.68E-02	6.05E-04	1.02E+00	9.89E+00	1.28E+00	2.90E+02	5.00E+02	3.16E+00	7.99E-01	2.17E-01
BX-102	3.85E+00	1.04E-01	1.01E-03	2.28E+00	6.04E+01	1.31E-05	2.70E+02	3.80E+03	4.28E-01	0.00E+00	1.01E+04
BX-108	3.32E-01	3.24E-02	8.88E-04	1.18E+00	1.37E+01	4.45E-01	3.45E+02	6.98E+02	3.49E+00	7.42E-02	2.87E-01
Sum of rows B103-B112 and BX102-BX111	4.36E+00	1.63E-01	2.50E-03	4.48E+00	8.40E+01	1.73E+00	9.05E+02	5.00E+03	7.07E+00	8.73E-01	1.01E+04
B-201	5.90E-10	1.32E-06	1.16E-10	1.19E-07	1.21E+00	1.75E+01	1.36E-04	2.93E+02	1.87E-12	0.00E+00	1.24E-04
B-203	2.42E-10	4.00E-07	2.01E-08	3.62E-08	3.81E-01	5.32E+00	3.79E-02	8.92E+01	5.68E-13	0.00E+00	2.43E-05
B-204	1.64E-09	5.38E-09	5.29E-07	1.28E-09	3.78E-01	3.98E-02	9.96E-01	9.43E+00	4.24E-15	0.00E+00	2.93E-05
Sum of rows B201-B204 and BX103-BX112	2.48E-09	1.73E-06	5.49E-07	1.57E-07	1.97E+00	2.29E+01	1.03E+00	3.92E+02	2.44E-12	0.00E+00	1.78E-04

[a] Field and Jones (2005)

[b] Corbin et al. (2005)

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Table C-34. Summary of Past Leak Inventory Estimates Based on the HTWOS [a] Model Assuming 10.2 kL (360 ft[3]) Remaining in the Tank I BY Tank Farm

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO₂ kg	NO₃ kg	NH₃ kg	Butanol kg	U kg
BY-107	3.51E-01	4.14E-02	1.27E-03	8.98E-01	1.31E+01	9.22E+00	5.32E+02	9.80E+02	4.13E+00	7.69E+00	1.41E-01
UPR-200-E-105	5.31E-01	7.04E-03	1.54E-04	1.76E-02	3.12E+01	3.26E+02	2.02E+02	9.34E+03	1.26E+02	0.00E+00	2.70E+00
Sum of rows BY101-BY110	8.82E-01	4.84E-02	1.43E-03	9.16E-01	4.43E+01	3.36E+02	7.34E+02	1.03E+04	1.30E+02	7.69E+00	2.84E+00
BY-108	1.17E-01	1.38E-02	4.24E-04	2.99E-01	4.36E+00	3.07E+00	1.77E+02	3.27E+02	1.38E+00	2.56E+00	4.70E-02
Sum of rows BY102-BY111	1.17E-01	1.38E-02	4.24E-04	2.99E-01	4.36E+00	3.07E+00	1.77E+02	3.27E+02	1.38E+00	2.56E+00	4.70E-02
BY-103	1.17E-01	1.38E-02	4.24E-04	2.99E-01	4.36E+00	3.07E+00	1.77E+02	3.27E+02	1.38E+00	2.56E+00	4.69E-02
UPR-200-E-110	1.18E-01	1.55E-03	3.40E-05	3.90E-03	6.90E+00	7.22E+01	4.47E+01	2.06E+03	2.78E+01	0.00E+00	5.95E-01
Sum of rows BY103-BY112	2.35E-01	1.53E-02	4.58E-04	3.03E-01	1.13E+01	7.52E+01	2.22E+02	2.39E+03	2.92E+01	2.56E+00	6.42E-01

[a] Kirkbride et al. (2005)

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C7.3 ANCILLARY EQUIPMENT (PIPELINES AND MUST) INVENTORY

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Table C-35. Summary of Final Pipeline Inventory Estimates for Waste Management Area B-BX-BY [a]

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO₂ kg	NO₃ kg	U kg
Pipeline	0.00E+00	5.93E-04	2.10E-06	1.28E-07	4.25E-03	0.00E+00	2.76E-02	2.55E-01	1.89E-02

[a] Lambert (2005)

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Table C-36. Summary of Extrapolated MUST Inventory for Waste Management Area B-BX-BY

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO₂ kg	NO₃ kg	U kg
244-BXR vault placed in tank row B101–B110	3.28E-03	1.11E-03	1.74E-04	7.03E-02	5.98E+00	8.47E+00	8.09E+00	7.09E+01	3.38E+01
241-BX-302A placed in tank row B101–B110	3.50E-04	1.19E-04	1.86E-05	7.50E-03	6.37E-01	9.04E-01	8.63E-01	7.56E+00	3.61E+00
241-B-301 placed in tank row B103-BX111	3.91E-04	1.37E-04	2.79E-06	7.05E-03	1.30E+00	1.14E+01	3.93E+00	3.46E+01	8.06E+00
244-BX-DCR placed in tank row B102-BX110	9.31E-04	3.15E-04	4.94E-05	1.99E-02	1.69E+00	2.40E+00	2.29E+00	2.01E+01	9.58E+00

1 **C7.4 POTENTIAL RETRIEVAL LEAKS**

Table C-37. Summary of 8,000-Gallon Retrieval Leak Inventory Estimates Based on HTWOS [a] Model Estimated Concentration for Retrieval Fluids for Single-Shell Tanks within B and BX Tank Farms

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO_2 kg	NO_3 kg	U kg
B-101	2.38E-02	1.69E-03	2.19E-05	4.08E-02	3.25E+00	2.18E+00	6.65E+02	2.37E+03	1.06E+01
B-104	9.22E-03	2.75E-03	1.12E-05	2.52E-01	7.33E-01	5.62E+01	6.87E+01	7.45E+03	5.27E-01
B-107	5.52E-02	1.13E-02	6.11E-04	5.96E-01	2.47E+00	3.56E+02	1.40E+02	6.24E+03	3.67E+00
B-110	5.77E-02	6.57E-02	3.81E-05	8.65E-01	3.38E+00	4.03E+01	5.25E+02	9.23E+03	5.35E-03
Sum of rows B101-B110	1.46E-01	8.14E-02	6.82E-04	1.75E+00	9.83E+00	4.55E+02	1.40E+03	2.53E+04	1.48E+01
B-102	2.09E-02	1.45E-03	2.32E-05	1.67E-02	5.35E+00	2.08E+02	1.05E+02	1.73E+03	5.02E-01
B-105	1.51E-02	1.17E-03	1.68E-05	2.74E-02	6.13E+00	2.93E+02	1.43E+02	1.84E+03	2.25E-02
B-108	1.73E-02	2.99E-03	4.18E-04	6.10E-03	1.41E+00	1.62E+02	1.23E+02	7.09E+02	2.88E-01
B-111	7.76E-02	7.84E-02	1.99E-05	5.22E+00	1.35E+01	3.02E+01	2.07E+03	3.69E+03	8.46E+00
BX-101	9.95E-03	2.10E-03	2.43E-04	1.08E-03	8.29E-01	1.91E+00	3.49E+01	2.62E+02	9.46E-01
BX-104	1.00E-01	7.74E-04	1.54E-04	8.47E-02	5.34E+00	2.55E+00	5.68E+01	1.15E+02	2.62E-02
BX-107	2.37E-03	2.08E-03	4.28E-06	2.63E-01	1.19E+00	3.31E+01	5.75E+01	9.69E+02	2.60E+00
BX-110	2.88E-01	3.98E-02	6.34E-03	5.51E-01	2.32E+01	1.12E+02	2.53E+02	5.95E+03	1.78E+00
Sum of rows B102-B111 and BX101-BX110	5.31E-01	1.29E-01	7.22E-03	6.17E+00	5.69E+01	8.43E+02	2.84E+03	1.53E+04	1.46E+01
B-103	1.61E-02	1.11E-03	1.76E-05	1.29E-02	4.65E+00	1.96E+02	9.20E+01	1.43E+03	1.41E+00
B-106	1.33E-02	1.79E-03	1.33E-05	1.42E-01	1.15E+00	4.50E+01	1.17E+02	2.91E+03	5.85E-01
B-109	1.54E-02	2.80E-03	4.02E-04	4.80E-03	7.43E+00	1.13E+02	4.53E+01	8.31E+02	1.11E+00
B-112	1.79E-01	2.47E-02	2.80E-03	5.23E-01	1.11E+01	2.34E+01	4.46E+02	1.82E+03	1.92E-01
BX-102	4.35E-03	1.04E-03	1.24E-04	1.48E-04	3.75E-01	6.59E-01	1.09E+01	3.38E+01	2.29E-01
BX-105	5.41E-01	5.93E-03	1.13E-04	1.08E-01	3.90E+01	2.12E+00	7.09E+01	1.54E+02	1.20E-01
BX-108	1.20E-02	1.33E-03	1.17E-05	9.22E-02	7.53E-01	1.75E+01	1.25E+02	1.73E+03	4.16E+00
BX-111	3.47E-01	3.23E-02	6.81E-03	3.04E-01	1.44E+01	5.63E+01	2.78E+02	6.10E+03	3.11E-01
Sum of rows B103-B112 and BX102-BX111	1.13E+00	7.10E-02	1.03E-02	1.19E+00	7.88E+01	4.54E+02	1.19E+03	1.50E+04	8.12E+00
BX-103	1.25E-01	3.33E-03	4.22E-04	1.79E-03	1.08E+00	2.52E+00	5.76E+01	4.32E+02	2.73E+00
BX-106	2.17E-01	2.48E-03	5.48E-05	1.22E-01	2.38E+01	1.73E+00	6.50E+01	1.59E+02	8.56E-01
BX-109	2.42E-02	1.05E-03	2.23E-05	6.21E-03	6.07E-01	4.71E+00	3.18E+02	3.46E+03	7.94E+00
BX-112	3.21E-03	1.59E-03	3.79E-06	1.57E-01	2.38E+00	1.52E+01	1.57E+02	4.34E+02	3.21E-02
Sum of rows BX103-BX112	3.69E-01	8.44E-03	5.03E-04	2.87E-01	2.78E+01	2.42E+01	5.98E+02	4.48E+03	1.16E+01

[a] Kirkbride et al. (2005)

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Table C-38. Summary of 8,000-Gallon Retrieval Leak Inventory Estimates Based on the HTWOS [a] Model Estimated Concentration for Retrieval Fluids for Single-Shell Tanks within BY Tank Farm

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO₂ kg	NO₃ kg	U kg
BY-101	3.72E-01	3.20E-02	7.48E-03	2.47E-01	1.27E+01	7.33E+01	1.38E+02	5.39E+03	1.27E+00
BY-104	3.74E-01	3.53E-02	6.99E-03	3.62E-01	2.63E+01	6.49E+01	6.98E+02	3.68E+03	3.54E-02
BY-107	3.30E-01	4.41E-02	8.12E-03	5.12E-01	1.40E+00	6.27E+01	7.07E+02	3.04E+03	9.00E-01
BY-110	3.50E-01	3.23E-02	6.76E-03	3.05E-01	4.60E+01	7.29E+01	6.55E+02	1.87E+03	7.42E+00
Sum of rows BY101-BY110	1.43E+00	1.44E-01	2.93E-02	1.43E+00	8.64E+01	2.74E+02	2.20E+03	1.40E+04	9.63E+00
BY-102	3.42E-01	4.71E-02	6.29E-03	8.28E-01	1.56E+01	2.02E+02	3.14E+02	1.57E+03	1.74E-01
BY-105	4.26E-01	3.82E-02	8.68E-03	9.87E-02	5.15E+00	3.08E+01	2.26E+02	7.57E+03	2.67E+00
BY-108	4.21E-01	5.62E-02	9.63E-03	7.00E-01	2.28E+00	1.58E+02	5.98E+02	3.54E+03	6.08E-02
BY-111	3.43E-01	4.58E-02	8.89E-03	5.03E-01	1.72E+01	1.40E+02	2.65E+02	2.21E+03	1.70E-01
Sum of rows BY102-BY111	1.53E+00	1.87E-01	3.35E-02	2.13E+00	4.03E+01	5.31E+02	1.40E+03	1.49E+04	3.07E+00
BY-103	3.80E-01	3.51E-02	7.50E-03	3.23E-01	1.78E+01	1.53E+02	1.75E+02	3.53E+03	4.88E-01
BY-106	3.68E-01	4.86E-02	1.13E-02	4.12E-01	6.40E+00	1.07E+02	4.87E+02	5.58E+03	8.54E-01
BY-109	3.78E-01	3.73E-02	6.83E-03	4.11E-01	1.27E+01	5.05E+02	5.84E+02	1.48E+03	4.26E-01
BY-112	2.71E-01	3.73E-02	5.42E-03	6.36E-01	9.62E+01	7.94E+01	3.47E+02	9.20E+02	6.85E-01
Sum of rows BY103-BY112	1.40E+00	1.58E-01	3.11E-02	1.78E+00	1.33E+02	8.44E+02	1.59E+03	1.15E+04	2.45E+00

[a] Kirkbride et al. (2005)

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1 **C8.0 WASTE MANAGEMENT AREA A-AX INVENTORY**

2 This section provides the inventory of the major risk-driving radionuclides and hazardous
3 chemicals found within WMA A-AX.

4 **C8.1 SINGLE-SHELL TANK RESIDUAL WASTE INVENTORY**

**Table C-39. Summary of Final Tank Residual Inventory Estimates Based on
HTWOS [a] Model Assuming 10.2 kL (360 ft^3) Remaining
in the Tank in A Tank Farm**

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO$_2$ kg	NO$_3$ kg	U kg
A-101	2.47E-02	8.27E-03	2.93E-04	2.88E-01	6.52E+02	1.24E+00	1.67E+02	3.08E+02	1.84E+02
A-102	3.70E-02	4.38E-03	3.70E-04	3.39E+00	2.92E+02	1.18E+00	2.56E+02	3.74E+02	1.19E+03
A-103	2.05E-02	5.41E-03	5.05E-04	1.34E+00	1.08E+02	2.47E+00	2.22E+02	3.16E+02	1.07E+02
Sum of rows A101-A103	8.22E-02	1.81E-02	1.17E-03	5.02E+00	1.05E+03	4.89E+00	6.45E+02	9.98E+02	1.48E+03
A-104	3.69E-03	7.63E-03	1.54E-05	3.10E-02	3.55E+00	5.74E-03	1.42E+01	1.36E+00	3.54E+01
A-105	8.90E-03	1.82E-02	9.10E-06	1.51E-01	1.89E+01	1.62E-01	3.58E+00	9.16E+01	1.25E-01
A-106	1.49E-02	1.60E-03	1.94E-04	2.22E-01	7.84E+01	1.77E-01	1.41E+02	1.55E+02	1.32E+01
Sum of rows A104-A106	2.75E-02	2.75E-02	2.19E-04	4.05E-01	1.01E+02	3.44E-01	1.59E+02	2.48E+02	4.88E+01

[a] Kirkbride et al. (2005)

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**Table C-40. Summary of Final Tank Residual Inventory Estimates Based on
the HTWOS [a] Model Assuming 10.2 kL (360 ft^3) Remaining
in the Tank in AX Tank Farm**

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO$_2$ kg	NO$_3$ kg	U kg
AX-101	2.76E-02	1.00E-02	3.30E-04	3.24E-01	5.98E+02	1.81E+00	1.85E+02	3.75E+02	1.69E+02
AX-103	2.49E-02	8.58E-03	3.25E-04	3.38E-01	2.81E+02	2.59E+00	2.59E+02	3.28E+02	2.97E+01
Sum of rows AX101-AX103	5.25E-02	1.86E-02	6.56E-04	6.62E-01	8.80E+02	4.41E+00	4.44E+02	7.03E+02	1.99E+02
AX-102	4.04E-02	1.45E-03	3.67E-04	1.93E-02	2.37E+01	1.29E+00	1.11E+02	5.62E+02	1.30E+02
AX-104	4.37E-03	1.25E-02	4.47E-06	1.14E+00	2.07E+00	6.21E-02	1.38E+00	2.81E+01	1.95E+01
Sum of rows AX102-AX104	4.48E-02	1.39E-02	3.72E-04	1.16E+00	2.58E+01	1.35E+00	1.12E+02	5.90E+02	1.49E+02

[a] Kirkbride et al. (2005)

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1   **C8.2   PAST TANK LEAKS AND UNPLANNED RELEASES INVENTORY**

**Table C-41. Summary of Past Leak Inventory Estimates within A Tank Farm
Based on Volume Estimates [a] and Soil Inventory Model [b]**

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO$_2$ kg	NO$_3$ kg	NH$_3$ kg	Butanol kg	U kg
A-103	6.33E-01	1.77E-01	5.29E-03	5.11E+00	5.99E+01	3.82E+01	2.59E+03	4.52E+03	3.24E+01	3.40E+01	6.45E-01
Sum of rows A101-A103	6.33E-01	1.77E-01	5.29E-03	5.11E+00	5.99E+01	3.82E+01	2.59E+03	4.52E+03	3.24E+01	3.40E+01	6.45E-01
A-104	8.05E-02	1.46E-02	2.91E-04	5.47E-01	6.71E+00	9.65E-02	1.66E+02	1.41E+02	2.27E+00	1.30E-02	2.34E-01
A-105	3.44E-02	1.02E-02	6.40E-05	5.12E-01	1.59E+00	1.42E-01	8.57E+01	3.35E+01	1.49E+00	4.75E-01	1.68E-01
Sum of rows A104-A106	1.15E-01	2.47E-02	3.55E-04	1.06E+00	8.31E+00	2.39E-01	2.52E+02	1.75E+02	3.76E+00	4.88E-01	4.02E-01

[a] Field and Jones (2005)

[b] Corbin et al. (2005)

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**Table C-42. Summary of Past Leak Inventory Estimates within AX Tank Farm
Based on Volume Estimates [a] and Soil Inventory Model [b]**

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO$_2$ kg	NO$_3$ kg	NH$_3$ kg	Butanol kg	U kg
AX-102	1.77E-01	1.19E-02	6.94E-05	8.04E-01	1.48E+00	1.39E-02	2.09E+01	1.95E+02	1.52E-01	1.65E-01	3.44E-01
Sum of rows AX102-AX104	1.77E-01	1.19E-02	6.94E-05	8.04E-01	1.48E+00	1.39E-02	2.09E+01	1.95E+02	1.52E-01	1.65E-01	3.44E-01

[a] Field and Jones (2005)

[b] Corbin et al. (2005)

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4   **C8.3   ANCILLARY EQUIPMENT (PIPELINES AND MUST) INVENTORY**

**Table C-43. Summary of Extrapolated MUST Inventory for
Waste Management Area A-AX**

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO$_2$ kg	NO$_3$ kg	U kg
241-A-350 placed in tank row A104-A106	2.54E-05	1.05E-05	3.21E-07	1.26E-03	2.67E-01	1.21E-03	1.86E-01	2.88E-01	3.54E-01
241-A-417 placed in tank row A104-A106	1.22E-03	5.08E-04	1.55E-05	6.05E-02	1.29E+01	5.84E-02	8.97E+00	1.39E+01	1.71E+01

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1 **C8.4 POTENTIAL RETRIEVAL LEAKS**

Table C-44. Summary of 8,000-Gallon Retrieval Leak Inventory Estimates Based on HTWOS [a] Model Estimated Concentration for Retrieval Fluids for Single-Shell Tanks within A Tank Farm

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO$_2$ kg	NO$_3$ kg	U kg
A-101	1.77E-01	5.93E-02	2.10E-03	2.07E+00	7.21E+00	8.91E+00	1.20E+03	2.21E+03	3.86E-03
A-102	2.67E-01	3.09E-02	2.67E-03	2.37E+00	4.11E+00	8.49E+00	1.84E+03	2.70E+03	9.85E-01
A-103	1.46E-01	3.74E-02	4.84E-04	2.11E+00	1.53E+00	1.03E+01	1.43E+03	2.14E+03	1.47E-01
Sum of rows A101-A103	5.91E-01	1.28E-01	5.25E-03	6.55E+00	1.29E+01	2.77E+01	4.47E+03	7.05E+03	1.14E+00
A-104	2.59E-02	3.73E-05	1.08E-04	2.17E-01	2.56E-01	4.03E-02	1.00E+02	9.54E+00	4.14E-06
A-105	6.31E-02	5.13E-03	6.44E-05	1.07E+00	1.06E+01	1.14E+00	2.54E+01	6.49E+02	1.93E-09
A-106	1.05E-01	1.13E-02	1.38E-03	1.57E+00	5.69E+00	1.25E+00	9.96E+02	1.10E+03	1.25E-01
Sum of rows A104-A106	1.94E-01	1.65E-02	1.55E-03	2.86E+00	1.66E+01	2.44E+00	1.12E+03	1.75E+03	1.25E-01

[a] Kirkbride et al. (2005)

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Table C-45. Summary of 8,000-Gallon Retrieval Leak Inventory Estimates Based on HTWOS [a] Model Estimated Concentration for Retrieval Fluids for Single-Shell Tanks within AX Tank Farm

Analyte→ Tank ↓	Tritium Ci	C-14 Ci	I-129 Ci	Tc-99 Ci	Cr kg	F kg	NO$_2$ kg	NO$_3$ kg	U kg
AX-101	1.98E-01	6.62E-02	2.38E-03	2.33E+00	2.22E+00	1.30E+01	1.33E+03	2.70E+03	1.26E-03
AX-103	1.79E-01	5.97E-02	2.33E-03	2.42E+00	1.06E+01	1.86E+01	1.86E+03	2.36E+03	5.71E-03
Sum of rows AX101-AX103	3.77E-01	1.26E-01	4.71E-03	4.75E+00	1.29E+01	3.16E+01	3.19E+03	5.05E+03	6.97E-03
AX-102	2.90E-01	1.04E-02	2.64E-03	1.39E-01	6.41E-01	9.27E+00	7.94E+02	4.03E+03	5.71E-03
AX-104	3.07E-02	7.29E-05	3.13E-05	8.02E+00	1.09E+00	4.36E-01	9.66E+00	1.97E+02	1.92E-08
Sum of rows AX102-AX104	3.21E-01	1.05E-02	2.67E-03	8.16E+00	1.74E+00	9.70E+00	8.03E+02	4.23E+03	5.71E-03

[a] Kirkbride et al. (2005)

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C9.0   REFERENCES

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APPENDIX D

PERFORMANCE ASSESSMENT RESULTS FOR

GROUNDWATER PATHWAY

CONTENTS

TABLES

TERMS

CAS   Chemical Abstracts Service
DMT   Decision Management Tool
K_d distribution coefficient
MUST   miscellaneous underground storage tank
SST   single-shell tank
STOMP   Subsurface Transport Over Multiple Phases
WMA   waste management area

1 **D1.0 INTRODUCTION**

2 This appendix contains a guide to the tabulated results generated during the groundwater
3 modeling analyses described in Chapter 4.0 of *Initial Single-Shell Tank System Performance*
4 *Assessment for the Hanford Site*. The tabulated results are contained in several Microsoft Excel[1]
5 spreadsheet files on the accompanying CD-ROM for the performance assessment. The
6 following detailed description will allow the user to understand how the spreadsheet files are
7 organized, and will assist the user in finding the groundwater results for specific source
8 components, specific sources, and specific contaminants. Chapter 4.0 only reported the results
9 for tank rows exhibiting the highest groundwater concentrations for each source component in a
10 given waste management area (WMA). The Excel spreadsheet files contain all the results
11 generated by the groundwater modeling analyses.

12 **D1.1 CD ROM FILE STRUCTURE**

13 The Excel spreadsheet files on the CD-ROM are organized by WMA. There is one folder
14 for each single-shell tank (SST) WMA and each folder is named using the letter designation for
15 the WMA (e.g., "C" for WMA C and "A-AX" for WMA A-AX). Within the folder, there are
16 three Excel files, one file for each source component. The files are named with the letter
17 designation for the WMA followed by an underscore and the name of the source component
18 (e.g., "A-AX_PastReleases" and "U_TankResiduals"). Table D-1 contains a list of all the
19 Excel files contained on the CD-ROM.

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[1] Microsoft and Excel are either registered trademarks or trademarks of Microsoft Corporation in the United States and/or other countries.

April 2006

**Table D-1. File Names and Locations of Single-Shell Tank
Performance Assessment Results**

Waste Management Area A-AX		
Source Component	**Folder**	**File Name**
Past releases	A-AX	A-AX_PastReleases.xls
Tank residuals	A-AX	A-AX_TankResiduals.xls
Ancillary equipment residuals	A-AX	A-AX_AncillaryResiduals.xls
Waste Management Area B-BX-BY		
Source Component	**Folder**	**File Name**
Past releases	B-BX-BY	B-BX-BY_PastReleases.xls
Tank residuals	B-BX-BY	B-BX-BY_TankResiduals.xls
Ancillary equipment residuals	B-BX-BY	B-BX-BY_AncillaryResiduals.xls
Waste Management Area C		
Source Component	**Folder**	**File Name**
Past releases	C	C_PastReleases.xls
Tank residuals	C	C_TankResiduals.xls
Ancillary equipment residuals	C	C_AncillaryResiduals.xls
Waste Management Area S-SX		
Source Component	**Folder**	**File Name**
Past releases	S-SX	S-SX_PastReleases.xls
Tank residuals	S-SX	S-SX_TankResiduals.xls
Ancillary equipment residuals	S-SX	S-SX_AncillaryResiduals.xls
Waste Management Area T		
Source Component	**Folder**	**File Name**
Past releases	T	T_PastReleases.xls
Tank residuals	T	T_TankResiduals.xls
Ancillary equipment residuals	T	T_AncillaryResiduals.xls
Waste Management Area TX-TY		
Source Component	**Folder**	**File Name**
Past releases	TX-TY	TX-TY_PastReleases.xls
Tank residuals	TX-TY	TX-TY_TankResiduals.xls
Ancillary equipment residuals	TX-TY	TX-TY_AncillaryResiduals.xls
Waste Management Area U		
Source Component	**Folder**	**File Name**
Past releases	U	U_PastReleases.xls
Tank residuals	U	U_TankResiduals.xls
Ancillary equipment residuals	U	U_AncillaryResiduals.xls

1

D1.2   SPREADSHEET ORGANIZATION

Each Excel spreadsheet file is separated into three groups of tabs:

- Row Definitions: One blue-colored tab named "RowDefinitions" that lists the sources that belong to each tank row.

- Individual Sources: A group of gray-colored tabs containing a tab for every source in the source component for the given WMA. Each tab contains the results for the source listed on the tab.

- Tank Row: A group of black-colored tabs containing the results for all the sources within a WMA tank row, when applicable. There will always be source tabs in a file, but there may be files that lack tank row tabs in the instance that there is only one source in a tank row.

The individual sources tabs use the following naming conventions: the name of the source, followed by an abbreviated source component name (i.e., Residuals or Leak), and, if applicable, the Subsurface Transport Over Multiple Phases (STOMP)[2] simulation used. For example, "AX-104ResidualCase02" is the AX-104 tank residuals with STOMP case 02 used to simulate contaminant transport.

The tank row tabs use the following naming conventions: the name of the tank row (as defined in the "RowDefinitions" tab), followed by an abbreviated source component name, and, if applicable, the STOMP simulation used. For example, "A-101RowResidualCase02" would be all the tank residuals in the tank row A-101 (defined in the "RowDefinitions" tab as Tanks A-101, A-102, and A-103) with STOMP Case 02 used to simulate contaminant transport.

For the tank residual source component, tank rows that have miscellaneous underground storage tanks (MUST) in them will have two spreadsheets: one spreadsheet will contain the results for just the SST residuals and will be named as described above; the second spreadsheet will contain the results for the SSTs and the MUSTs that are in the same row and will be named the same as above with "+MUSTs" appended to the end of the name (e.g., A-104RowResidualCase02+MUSTs).

Each spreadsheet in the individual source and tank row groups has an identical arrangement because the spreadsheets are automatically populated and formatted using the report generation feature of the Decision Management Tool©[3] Table D-2 lists the standard columns found in each spreadsheet.

The spreadsheets only report values and results for analytes that were included in the selected source inventories.

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[2] Subsurface Transport Over Multiple Phases (STOMP) is copyrighted by Battelle Memorial Institute, 1996.
[3] Copyright pending.

Table D-2. Excel File Column Header Explanation

Column Number	Column Heading	Description
A	CAS Number	The CAS Number is a unique chemical identifier established by the Chemical Abstracts Service of the American Chemical Society. The CAS number is used to identify analytes in the DMT.
B	Analyte Name	The Analyte Name provided here is a common name of the analyte to make identification easier for the user.
C	Cumulative Inventory	This is the inventory of the analyte found in column B of the same row. It is a summation of the inventories of all the selected sources (see column I for a list of all selected sources).
D	Concentration	This is the peak fenceline concentration for the cumulative inventory reported in column C of the same row.
E	Peak Year	This is the year in which the peak fenceline concentration reported in column D of the same row occurs.
F	K_d	K_d is an empirically derived chemical parameter describing the mobility of a chemical through a given media. The K_d values used in this analysis describe analyte migration through the geochemical environment found in a typical tank farm vadose zone. In this analysis, K_d values of 0 mL/g are mobile, 0.2 mL/g are semi-mobile, and 3 mL/g are less-mobile.
G	K_d Bin	STOMP simulations model the migration of generic analytes with eight discreet K_d values. These K_d values are known as bins. The DMT matches the K_d of a selected analyte to the closest K_d bin in the STOMP run to use in projecting groundwater concentrations for that analyte. This column displays the K_d bin selected by the DMT.
H	Half-Life	This is the time required for the quantity of an analyte to decay to half its initial value. Note that chemical half-lives are not considered in this analysis.
I, J, K	None	The box found at the top of columns I, J, and K is a DMT-generated set of information about the parameters used to generate the report. The box gives the time and date the report was generated, the user who generated the report, and the version of the DMT that was used to generate the report. This box also lists the sources and related inventories and STOMP transport simulations selected to generate the report. Column I will list the source, column J will list the inventory or inventories applied to the source in column I, and column K will list the STOMP runs applicable to the inventories in column J that were selected for the analysis. Other information provided about the report parameters are: dilution factor, which is the two-dimensional (2D) to three-dimensional (3D) factor applied to the results; and compliance monitoring start year, which is the first year in the simulation for which results are recorded.

CAS = Chemical Abstracts Service

K_d = distribution coefficient

1

APPENDIX E
TANK BY TANK RESULTS FOR INTRUDER
AND AIR PATHWAYS

1 **CONTENTS**

6
7
8
9 **TABLES**

15

E1.0 WASTE DIMENSIONS AND FRACTIONS EXHUMED

This appendix contains detailed intruder and air pathway analyses information based on residual wastes remaining in the tanks after closure. Waste dimensions are needed to calculate the fraction of the total waste left in the various underground tanks and unplanned releases that might be brought to the surface by an intruder. For purposes of estimating the fraction of waste exhumed, the waste falls into two categories: 1) the residual waste in the underground tanks and 2) the contaminated soil from leaks and other unplanned releases.

The residual waste in the tanks is estimated using the tank dimensions together with the maximum waste volume that may be left in the tanks. The tank inside diameter is either 75 ft or 20 ft. The corresponding residual waste volume limits from the *Hanford Federal Facility Agreement and Consent Order* (Ecology et al. 1989) are 360 ft^3 (100-Series tanks) and 30 ft^3 (200-Series tanks). The average residual tank waste encountered by the well driller is a cylinder with a vertical axis. The bounding cylinder diameter is either 75 ft or 20 ft and the height of the cylinder is either 0.978 in. or 1.146 in., assuming the residual waste is uniformly spread across the bottom of the tank. The dimensions used in the intruder and air pathway calculations are listed in Table E-1.

Table E-1. Dimensions for Residual Waste in Underground Storage Tanks

Location	Waste Volume	Cylinder Diameter	Cylinder Height	Horizontal Area
100-Series tanks	360 ft^3 or 2,693 gal (10,190 L)	75.0 ft (22.86 m)	0.978 in. (2.484 cm)	4,418 ft^2 (410 m^2)
C-106	370 ft^3 or 2,771 gal (10,488 L)	75 ft (22.86 m)	1.006 in. (2.56 cm)	4,418 ft^2 (410.4 m^2)
200-Series tanks	30 ft^3 or 224.4 gal (850 L)	20 ft (6.10 m)	1.15 in. (2.91 cm)	314 ft^2 (29.2 m^2)

The average density of the 100-Series tanks residual waste is 1.42 kg/L. This density was used for all the other tank residuals shown in Table E-1 (except tank C-106). Unplanned releases to the soil use the bulk density of the soil, 1.7 kg/L. The average density of the soil in the borehole is estimated to be 1.7 kg/L based on the porosity of Hanford formation sand presented in a recent risk assessment for radioactive waste disposal in the 200 East Area, *Risk Assessment Supporting the Decision on the Initial Selection of Supplemental ILAW Technologies* (Mann et al. 2003).

The unplanned soil contamination is also represented as cylinders with vertical axes. Two important parameters used to represent this residual soil contamination are the cylinder volume and the relative width and height of the cylinder.

The volume of the cylinder is the estimated liquid leak volume divided by the assumed soil filling fraction, 10%. This average filling fraction is the volume of waste per unit volume of soil. For comparison, the pore fraction (the available volume per unit volume of soil) near the tanks ranges from 5% to 50% (Khaleel and Freeman 1995). In addition, the residual water fractions

1 listed in Khaleel and Freeman (1995) range from 0% to 15%. The number chosen (10%) is a
2 representative value that may be somewhat larger than it needs to be.

3 The relative width and height of the cylinder depends on the relative amounts of horizontal and
4 vertical migration in the vadose zone. The numerical parameter will be referred to as the
5 anisotropy. Plume anisotropy is the ratio of the horizontal dimension to the vertical dimension of
6 the plume. When there is enough liquid to saturate the soil, the liquid tends to move downward,
7 giving a fractional plume anisotropy. Under unsaturated conditions, the plume spreads
8 horizontally as well as vertically, leading to anisotropies greater than 1. Some of the observed
9 anisotropy is the result of soil discontinuities created at the time the underground tanks were
10 constructed. Horizontal spreading occurs at this boundary. An example was analyzed in
11 Routson et al. (1979) for the large leak from 241-T-106. For this leak, the observed anisotropy is
12 about 2. For the intruder calculations, the value 1.0 was chosen as a representative number.
13 Larger anisotropies translate into smaller intruder doses because the well intercepts less
14 contaminated soil.

15 The equations used to calculate the cylinder dimensions from the volume leaked, the filling
16 fraction, and the anisotropy are listed below. The liquid volumes released and the resulting
17 plume dimensions are listed in Table E-2.

$$\text{Soil Volume} = \frac{V}{\theta} = \frac{\pi}{4}D^2 L = \frac{\pi}{4\alpha}D^3$$

18   **Eq. E-1**

$$\text{thus,} \quad D = \left(\frac{4\alpha V}{\pi\theta}\right)^{1/3} \quad \text{and} \quad L = \frac{D}{\alpha}$$

19 where:
20 D = cylinder diameter, in meters
21 L = cylinder height, in meters
22 V = volume of aqueous waste that leaks into the soil, in cubic meters
23 α = cylinder anisotropy, the ratio of the cylinder diameter to the cylinder height, 1.00
24 π = 3.14159265358979 . . .
25 θ = filling fraction of the soil, the average ratio of waste liquid volume to soil volume, 10%.

Table E-2. Dimensions for Waste in the Unplanned Releases (4 pages)

Location	Liquid Waste Volume	Soil Volume	Cylinder Diameter	Cylinder Height	Horizontal Area
A-103 leak	5,500 gal (20,820 L)	7,352 ft^3 (208 m^3)	21.1 ft (6.42 m)	21.1 ft (6.42 m)	349 ft^2 (32.4 m^2)
A-104 leak	2,000 gal (7,571 L)	2,674 ft^3 (75.71 m^3)	15.0 ft (4.59 m)	15.0 ft (4.59 m)	177.7 ft^2 (16.51 m^2)
A-105 leak	1,000 gal (3,785 L)	1,337 ft^3 (37.85 m^3)	11.9 ft (3.64 m)	11.9 ft (3.64 m)	112.0 ft^2 (10.40 m^2)
AX-102 leak	3,000 gal (11,356 L)	4,010 ft^3 (114 m^3)	17.2 ft (5.25 m)	17.2 ft (5.25 m)	233 ft^2 (21.6 m^2)
B-107 leak	14,000 gal (52,996 L)	18,715 ft^3 (530 m^3)	28.8 ft (8.77 m)	28.8 ft (8.77 m)	650 ft^2 (60.4 m^2)
B-110 leak	10,000 gal (37,854 L)	13,368 ft^3 (379 m^3)	25.7 ft (7.84 m)	25.7 ft (7.84 m)	520 ft^2 (48.3 m^2)
B-112 leak	2,000 gal (7,571 L)	2,674 ft^3 (75.7 m^3)	15.0 ft (4.59 m)	15.0 ft (4.59 m)	178 ft^2 (16.5 m^2)
B-201 leak	1,200 gal (4,542 L)	1,604 ft^3 (45.4 m^3)	12.7 ft (3.87 m)	12.7 ft (3.87 m)	126 ft^2 (11.7 m^2)
B-203 leak	300 gal (1,136 L)	401 ft^3 (11.4 m^3)	7.99 ft (2.44 m)	7.99 ft (2.44 m)	50.2 ft^2 (4.66 m^2)
B-204 leak	400 gal (1,514 L)	535 ft^3 (15.1 m^3)	8.80 ft (2.68 m)	8.80 ft (2.68 m)	60.8 ft^2 (5.65 m^2)
UPR-200-E-6	1,000 gal (3,785 L)	1,337 ft^3 (37.9 m^3)	11.9 ft (3.64 m)	11.9 ft (3.64 m)	112 ft^2 (10.4 m^2)
UPR-200-E-73	92 gal (348 L)	123 ft^3 (3.48 m^3)	5.39 ft (1.64 m)	5.39 ft (1.64 m)	22.8 ft^2 (2.12 m^2)
UPR-200-E-74	10 gal (37.9 L)	13.4 ft^3 (0.379 m^3)	2.57 ft (0.784 m)	2.57 ft (0.784 m)	5.20 ft^2 (0.483 m^2)
UPR-200-E-75	2,000 gal (7,571 L)	2,674 ft^3 (75.7 m^3)	15.0 ft (4.59 m)	15.0 ft (4.59 m)	178 ft^2 (16.5 m^2)
UPR-200-E-108	196 gal (742 L)	262 ft^3 (7.42 m^3)	6.94 ft (2.11 m)	6.94 ft (2.11 m)	37.8 ft^2 (3.51 m^2)
UPR-200-E-109	150 gal (568 L)	201 ft^3 (5.68 m^3)	6.34 ft (1.93 m)	6.34 ft (1.93 m)	31.6 ft^2 (2.94 m^2)
UPR-200-E-38	5,400 gal (20,441 L)	7,219 ft^3 (204 m^3)	20.9 ft (6.38 m)	20.9 ft (6.38 m)	345 ft^2 (32.0 m^2)
BX-101 leak	4,000 gal (15,142 L)	5,347 ft^3 (151 m^3)	19.0 ft (5.78 m)	19.0 ft (5.78 m)	282 ft^2 (26.2 m^2)
BX-102 leak	91,600 gal (346,744 L)	122,451 ft^3 (3,467 m^3)	53.8 ft (16.4 m)	53.8 ft (16.4 m)	2,275 ft^2 (211.4 m^2)

Table E-2. Dimensions for Waste in the Unplanned Releases (4 pages)

Location	Liquid Waste Volume	Soil Volume	Cylinder Diameter	Cylinder Height	Horizontal Area
BX-108 leak	2,500 gal (9,464 L)	3,342 ft^3 (94.6 m^3)	16.2 ft (4.94 m)	16.2 ft (4.94 m)	206 ft^2 (19.2 m^2)
BY-103 leak	400 gal (1,514 L)	535 ft^3 (15.1 m^3)	8.80 ft (2.68 m)	8.80 ft (2.68 m)	60.8 ft^2 (5.65 m^2)
BY-107 leak	1,200 gal (4,542 L)	1,604 ft^3 (45.4 m^3)	12.7 ft (3.87 m)	12.7 ft (3.87 m)	126 ft^2 (11.7 m^2)
BY-108 leak	400 gal (1,514 L)	535 ft^3 (15.1 m^3)	8.80 ft (2.68 m)	8.80 ft (2.68 m)	60.8 ft^2 (5.65 m^2)
UPR-200-E-105	23,000 gal (87,064 L)	30,747 ft^3 (871 m^3)	34.0 ft (10.3 m)	34.0 ft (10.3 m)	906 ft^2 (84.1 m^2)
UPR-200-E-110	5,100 gal (19,306 L)	6,818 ft^3 (193 m^3)	20.6 ft (6.26 m)	20.6 ft (6.26 m)	332 ft^2 (30.8 m^2)
C-101 leak	1,000 gal (3,785 L)	1,337 ft^3 (37.9 m^3)	11.9 ft (3.64 m)	11.9 ft (3.64 m)	112 ft^2 (10.4 m^2)
C-105 leak	1,000 gal (3,785 L)	1,337 ft^3 (37.9 m^3)	11.9 ft (3.64 m)	11.9 ft (3.64 m)	112 ft^2 (10.4 m^2)
C-110 leak	2,000 gal (7,571 L)	2,674 ft^3 (75.7 m^3)	15.0 ft (4.59 m)	15.0 ft (4.59 m)	178 ft^2 (16.5 m^2)
C-111 leak	5,500 gal (20,820 L)	7,352 ft^3 (208 m^3)	21.1 ft (6.42 m)	21.1 ft (6.42 m)	349 ft^2 (32.4 m^2)
C-201 leak	550 gal (2,082 L)	735 ft^3 (20.8 m^3)	9.78 ft (2.98 m)	9.78 ft (2.98 m)	75.2 ft^2 (6.98 m^2)
C-202 leak	450 gal (1,703 L)	602 ft^3 (17.0 m^3)	9.15 ft (2.79 m)	9.15 ft (2.79 m)	65.7 ft^2 (6.11 m^2)
C-203 leak	400 gal (1,514 L)	535 ft^3 (15.1 m^3)	8.80 ft (2.68 m)	8.80 ft (2.68 m)	60.8 ft^2 (5.65 m^2)
C-204 leak	350 gal (1,325 L)	468 ft^3 (13.2 m^3)	8.41 ft (2.56 m)	8.41 ft (2.56 m)	55.6 ft^2 (5.17 m^2)
UPR-200-E-107	5 gal (18.9 L)	6.68 ft^3 (0.189 m^3)	2.04 ft (0.622 m)	2.04 ft (0.622 m)	3.27 ft^2 (0.304 m^2)
UPR-200-E-81	36,000 gal (136,275 L)	48,125 ft^3 (1,363 m^3)	39.4 ft (12.0 m)	39.4 ft (12.0 m)	1,221 ft^2 (113.4 m^2)
UPR-200-E-82	2,600 gal (9,842 L)	3,476 ft^3 (98.4 m^3)	16.4 ft (5.00 m)	16.4 ft (5.00 m)	212 ft^2 (19.7 m^2)
UPR-200-E-86	18,500 gal (70,030 L)	24,731 ft^3 (700 m^3)	31.6 ft (9.62 m)	31.6 ft (9.62 m)	783 ft^2 (72.8 m^2)
S-104 leak	24,000 gal (90,850 L)	32,083 ft^3 (908 m^3)	34.4 ft (10.5 m)	34.4 ft (10.5 m)	932 ft^2 (86.5 m^2)

Table E-2. Dimensions for Waste in the Unplanned Releases (4 pages)

Location	Liquid Waste Volume	Soil Volume	Cylinder Diameter	Cylinder Height	Horizontal Area
SX-104 leak	6,000 gal (22,712 L)	8,021 ft^3 (227 m^3)	21.7 ft (6.61 m)	21.7 ft (6.61 m)	370 ft^2 (34.3 m^2)
SX-107 leak	15,000 gal (56,781 L)	20,052 ft^3 (568 m^3)	29.4 ft (8.98 m)	29.4 ft (8.98 m)	681 ft^2 (63.3 m^2)
SX-108 leak	35,000 gal (132,489 L)	46,788 ft^3 (1,325 m^3)	39.1 ft (11.9 m)	39.1 ft (11.9 m)	1,198 ft^2 (111.3 m^2)
SX-109 leak	2,000 gal (7,571 L)	2,674 ft^3 (75.7 m^3)	15.0 ft (4.59 m)	15.0 ft (4.59 m)	178 ft^2 (16.5 m^2)
SX-110 leak	1,000 gal (3,785 L)	1,337 ft^3 (37.9 m^3)	11.9 ft (3.64 m)	11.9 ft (3.64 m)	112 ft^2 (10.4 m^2)
SX-111 leak	500 gal (1,893 L)	668 ft^3 (18.9 m^3)	9.48 ft (2.89 m)	9.48 ft (2.89 m)	70.5 ft^2 (6.55 m^2)
SX-112 leak	1,000 gal (3,785 L)	1,337 ft^3 (37.9 m^3)	11.9 ft (3.64 m)	11.9 ft (3.64 m)	112 ft^2 (10.4 m^2)
SX-113 leak	15,000 gal (56,781 L)	20,052 ft^3 (568 m^3)	29.4 ft (8.98 m)	29.4 ft (8.98 m)	681 ft^2 (63.3 m^2)
SX-115 leak	50,000 gal (189,271 L)	66,840 ft^3 (1,893 m^3)	44.0 ft (13.4 m)	44.0 ft (13.4 m)	1,520 ft^2 (141.2 m^2)
T-101 leak	10,000 gal (37,854 L)	13,368 ft^3 (379 m^3)	25.7 ft (7.84 m)	25.7 ft (7.84 m)	520 ft^2 (48.3 m^2)
T-103 leak	3,000 gal (11,356 L)	4,010 ft^3 (114 m^3)	17.2 ft (5.25 m)	17.2 ft (5.25 m)	233 ft^2 (21.6 m^2)
T-106 leak	115,000 gal (435,322 L)	153,733 ft^3 (4,353 m^3)	58.1 ft (17.7 m)	58.1 ft (17.7 m)	2,648 ft^2 (246.0 m^2)
T-108 leak	1,000 gal (3,785 L)	1,337 ft^3 (37.9 m^3)	11.9 ft (3.64 m)	11.9 ft (3.64 m)	112 ft^2 (10.4 m^2)
T-109 leak	1,000 gal (3,785 L)	1,337 ft^3 (37.9 m^3)	11.9 ft (3.64 m)	11.9 ft (3.64 m)	112 ft^2 (10.4 m^2)
T-111 leak	1,000 gal (3,785 L)	1,337 ft^3 (37.9 m^3)	11.9 ft (3.64 m)	11.9 ft (3.64 m)	112 ft^2 (10.4 m^2)
TX-107 leak	8,000 gal (30,283 L)	10,694 ft^3 (303 m^3)	23.9 ft (7.28 m)	23.9 ft (7.28 m)	448 ft^2 (41.6 m^2)
UPR-200-W-100	2,500 gal (9,464 L)	3,342 ft^3 (94.6 m^3)	16.2 ft (4.94 m)	16.2 ft (4.94 m)	206 ft^2 (19.2 m^2)
UPR-200-W-12	5 gal (18.9 L)	6.68 ft^3 (0.189 m^3)	2.04 ft (0.622 m)	2.04 ft (0.622 m)	3.27 ft^2 (0.304 m^2)
TY-101 leak	1,000 gal (3,785 L)	1,337 ft^3 (37.9 m^3)	11.9 ft (3.64 m)	11.9 ft (3.64 m)	112 ft^2 (10.4 m^2)

Table E-2. Dimensions for Waste in the Unplanned Releases (4 pages)

Location	Liquid Waste Volume	Soil Volume	Cylinder Diameter	Cylinder Height	Horizontal Area
TY-103 leak	3,000 gal (11,356 L)	4,010 ft^3 (114 m^3)	17.2 ft (5.25 m)	17.2 ft (5.25 m)	233 ft^2 (21.6 m^2)
TY-104 leak	1,400 gal (5,300 L)	1,872 ft^3 (53.0 m^3)	13.4 ft (4.07 m)	13.4 ft (4.07 m)	140 ft^2 (13.0 m^2)
TY-105 leak	35,000 gal (132,489 L)	46,788 ft^3 (1,325 m^3)	39.1 ft (11.9 m)	39.1 ft (11.9 m)	1,198 ft^2 (111.3 m^2)
TY-106 leak	20,000 gal (75,708 L)	26,736 ft^3 (757 m^3)	32.4 ft (9.88 m)	32.4 ft (9.88 m)	825 ft^2 (76.6 m^2)
U-101 leak	5,000 gal (18,927 L)	6,684 ft^3 (189 m^3)	20.4 ft (6.22 m)	20.4 ft (6.22 m)	327 ft^2 (30.4 m^2)
U-104 leak	55,000 gal (208,198 L)	73,524 ft^3 (2,082 m^3)	45.4 ft (13.8 m)	45.4 ft (13.8 m)	1,619 ft^2 (150.4 m^2)
U-110 leak	6,500 gal (24,605 L)	8,689 ft^3 (246 m^3)	22.3 ft (6.79 m)	22.3 ft (6.79 m)	390 ft^2 (36.2 m^2)
U-112 leak	8,500 gal (32,176 L)	11,363 ft^3 (322 m^3)	24.4 ft (7.43 m)	24.4 ft (7.43 m)	466 ft^2 (43.3 m^2)
UPR-200-W-132	500 gal (1,893 L)	668 ft^3 (18.9 m^3)	9.48 ft (2.89 m)	9.48 ft (2.89 m)	70.5 ft^2 (6.55 m^2)
UPR-200-W-24	36 gal (136 L)	48.1 ft^3 (1.36 m^3)	3.94 ft (1.20 m)	3.94 ft (1.20 m)	12.2 ft^2 (1.13 m^2)

As a final note, the fraction of the waste that is exhumed is the square of the ratio of the well diameter to the waste diameter. The formula is shown below as Equation E-2. The fractions for the well diameters used in the various intruder scenarios are listed in Table E-3.

$$\text{Fraction Exhumed} = \left(\frac{d}{D}\right)^2 = d^2\left(\frac{\pi\,\theta}{4\,\alpha\,V}\right)^{2/3} \qquad \textbf{Eq. E-2}$$

where:

d = borehole diameter, in meters
D = cylinder diameter, in meters
L = cylinder height, in meters
V = volume of aqueous waste that leaks into the soil, in cubic meters
α = cylinder anisotropy, the ratio of the cylinder diameter to the cylinder height, 1.00
π = 3.14159265358979 . . .
θ = filling fraction of the soil, the average ratio of waste liquid volume to soil volume, 10%.

Table E-3. Borehole Dimensions and Fractions Exhumed (3 pages)

Item	Suburban Garden	Rural Pasture	Commercial Farm
Borehole diameter	6.50 in. (16.51 cm)	10.50 in. (26.67 cm)	16.50 in. (41.91 cm)
Hole area	214 cm^2	559 cm^2	1,380 cm^2
100-Series tanks	5.22E-05	1.36E-04	3.36E-04
200-Series tanks	7.34E-04	1.91E-03	4.73E-03
A-103 leak	6.61E-04	1.72E-03	4.26E-03
A-104 leak	1.30E-03	3.38E-03	8.35E-03
A-105 leak	2.06E-03	5.37E-03	1.33E-02
AX-102 leak	9.89E-04	2.58E-03	6.38E-03
B-107 leak	3.54E-04	9.25E-04	2.28E-03
B-110 leak	4.43E-04	1.16E-03	2.86E-03
B-112 leak	1.30E-03	3.38E-03	8.35E-03
B-201 leak	1.82E-03	4.76E-03	1.17E-02
B-203 leak	4.59E-03	1.20E-02	2.96E-02
B-204 leak	3.79E-03	9.89E-03	2.44E-02
UPR-200-E-6	2.06E-03	5.37E-03	1.33E-02
UPR-200-E-73	1.01E-02	2.64E-02	6.51E-02
UPR-200-E-74	4.43E-02	1.16E-01	2.86E-01
UPR-200-E-75	1.30E-03	3.38E-03	8.35E-03
UPR-200-E-108	6.10E-03	1.59E-02	3.93E-02
UPR-200-E-109	7.29E-03	1.90E-02	4.70E-02
UPR-200-E-38	6.69E-04	1.74E-03	4.31E-03
BX-101 leak	8.17E-04	2.13E-03	5.26E-03
BX-102 leak	1.01E-04	2.64E-04	6.53E-04
BX-108 leak	1.12E-03	2.92E-03	7.20E-03
BY-103 leak	3.79E-03	9.89E-03	2.44E-02
BY-107 leak	1.82E-03	4.76E-03	1.17E-02
BY-108 leak	3.79E-03	9.89E-03	2.44E-02
UPR-200-E-105	2.54E-04	6.64E-04	1.64E-03
UPR-200-E-110	6.95E-04	1.81E-03	4.48E-03

Table E-3. Borehole Dimensions and Fractions Exhumed (3 pages)

Item	Suburban Garden	Rural Pasture	Commercial Farm
C-101 leak	2.06E-03	5.37E-03	1.33E-02
C-105 leak	2.06E-03	5.37E-03	1.33E-02
C-110 leak	1.30E-03	3.38E-03	8.35E-03
C-111 leak	6.61E-04	1.72E-03	4.26E-03
C-201 leak	3.07E-03	8.00E-03	1.98E-02
C-202 leak	3.50E-03	9.15E-03	2.26E-02
C-203 leak	3.79E-03	9.89E-03	2.44E-02
C-204 leak	4.14E-03	1.08E-02	2.67E-02
UPR-200-E-107	7.04E-02	1.84E-01	4.54E-01
UPR-200-E-81	1.89E-04	4.93E-04	1.22E-03
UPR-200-E-82	1.09E-03	2.84E-03	7.01E-03
UPR-200-E-86	2.94E-04	7.68E-04	1.90E-03
S-104 leak	2.47E-04	6.45E-04	1.59E-03
SX-104 leak	6.23E-04	1.63E-03	4.02E-03
SX-107 leak	3.38E-04	8.83E-04	2.18E-03
SX-108 leak	1.92E-04	5.02E-04	1.24E-03
SX-109 leak	1.30E-03	3.38E-03	8.35E-03
SX-110 leak	2.06E-03	5.37E-03	1.33E-02
SX-111 leak	3.27E-03	8.53E-03	2.11E-02
SX-112 leak	2.06E-03	5.37E-03	1.33E-02
SX-113 leak	3.38E-04	8.83E-04	2.18E-03
SX-115 leak	1.52E-04	3.96E-04	9.77E-04
T-101 leak	4.43E-04	1.16E-03	2.86E-03
T-103 leak	9.89E-04	2.58E-03	6.38E-03
T-106 leak	8.70E-05	2.27E-04	5.61E-04
T-108 leak	2.06E-03	5.37E-03	1.33E-02
T-109 leak	2.06E-03	5.37E-03	1.33E-02
T-111 leak	2.06E-03	5.37E-03	1.33E-02
TX-107 leak	5.15E-04	1.34E-03	3.32E-03
UPR-200-W-100	1.12E-03	2.92E-03	7.20E-03
UPR-200-W-12	7.04E-02	1.84E-01	4.54E-01

Table E-3. Borehole Dimensions and Fractions Exhumed (3 pages)

Item	Suburban Garden	Rural Pasture	Commercial Farm
TY-101 leak	2.06E-03	5.37E-03	1.33E-02
TY-103 leak	9.89E-04	2.58E-03	6.38E-03
TY-104 leak	1.64E-03	4.29E-03	1.06E-02
TY-105 leak	1.92E-04	5.02E-04	1.24E-03
TY-106 leak	2.79E-04	7.29E-04	1.80E-03
U-101 leak	7.04E-04	1.84E-03	4.54E-03
U-104 leak	1.42E-04	3.71E-04	9.17E-04
U-110 leak	5.91E-04	1.54E-03	3.81E-03
U-112 leak	4.94E-04	1.29E-03	3.18E-03
UPR-200-W-132	3.27E-03	8.53E-03	2.11E-02
UPR-200-W-24	1.89E-02	4.93E-02	1.22E-01

1 **E2.0   INTRUDER DOSES FOR ALL WASTE ITEMS**

2 Table E-4 shows the doses at 500 years after closure for the tank residuals and releases to soil in

3 each waste management area. Several tanks have experienced significant leaks into the soil.

4 An intruder well could pass through the tank and miss the leak plume, or pass through the leak

5 plume and miss the tank, or hit both. Doses for all three combinations are listed in the table.

6 The dose from the tank alone is listed as the tank number without adjectives. The dose from the

7 soil plume alone is listed as the tank number followed by the word "Leak." The dose from a well

8 that passes through both the tank residual and tank leak plume is shown as the tank number

9 followed by the words "Tank + Leak." In Table E-4, **bold** text indicates that the calculated dose

10 exceeds the performance objective for the scenario (500 mrem/yr for well driller and

11 100 mrem/yr for post-intruder resident).

Table E-4. Intruder Dose (mrem) for All Tanks and Leaks (10 pages)

Source	Well Driller mrem	Suburban Garden mrem	Rural Pasture mrem	Commercial Farm mrem
A-101	2.01E+00	1.19E+01	5.38E-01	1.52E-02
A-102	5.48E+00	3.48E+01	1.54E+00	4.30E-02
A-103	6.85E-01	4.58E+00	1.94E-01	5.31E-03
A-103 Leak	6.17E-01	1.89E+01	3.35E-01	3.22E-03
A-103 Tank + Leak	1.30E+00	2.35E+01	5.29E-01	8.53E-03
A-104	2.62E+00	1.67E+01	7.64E-01	2.07E-02
A-104 Leak	2.28E-01	4.51E+00	9.66E-02	1.36E-03
A-104 Tank + Leak	2.85E+00	2.12E+01	8.61E-01	2.20E-02
A-105	9.45E+00	5.75E+01	2.63E+00	7.32E-02
A-105 Leak	2.69E-01	6.22E+00	1.23E-01	1.45E-03
A-105 Tank + Leak	9.72E+00	6.37E+01	2.75E+00	7.47E-02
A-106	2.29E+00	1.43E+01	6.45E-01	1.81E-02
AX-101	2.21E+00	1.25E+01	5.69E-01	1.60E-02
AX-102	1.79E+01	**1.03E+02**	4.69E+00	1.32E-01
AX-102 Leak	2.45E-01	4.99E+00	1.01E-01	1.45E-03
AX-102 Tank + Leak	1.81E+01	**1.08E+02**	4.79E+00	1.34E-01
AX-103	4.33E+00	2.52E+01	1.15E+00	3.21E-02
AX-104	5.61E+00	3.40E+01	1.56E+00	4.19E-02
B-101	4.19E+00	2.29E+01	1.03E+00	2.90E-02
B-102	7.53E-02	4.12E-01	1.85E-02	5.20E-04
B-103	3.90E-01	2.13E+00	9.59E-02	2.70E-03

Table E-4. Intruder Dose (mrem) for All Tanks and Leaks (10 pages)

Source	Well Driller mrem	Suburban Garden mrem	Rural Pasture mrem	Commercial Farm mrem
B-104	2.23E-01	1.23E+00	5.50E-02	1.54E-03
B-105	1.18E-01	6.49E-01	2.91E-02	8.18E-04
B-106	1.09E-01	6.16E-01	2.74E-02	7.46E-04
B-107	9.52E-02	5.44E-01	2.37E-02	6.54E-04
B-107 Leak	2.10E-01	1.59E+00	5.87E-02	1.38E-03
B-107 Tank + Leak	3.05E-01	2.14E+00	8.24E-02	2.03E-03
B-108	1.12E-02	4.64E-02	2.37E-03	6.59E-05
B-109	5.06E-02	2.79E-01	1.22E-02	3.33E-04
B-110	3.70E-01	2.05E+00	9.16E-02	2.54E-03
B-110 Leak	5.88E-02	1.39E+00	2.57E-02	2.85E-04
B-110 Tank + Leak	4.29E-01	3.44E+00	1.17E-01	2.83E-03
B-111	2.78E-01	1.81E+00	7.38E-02	1.91E-03
B-112	1.12E-02	7.66E-02	2.57E-03	6.06E-05
B-112 Leak	3.22E-01	7.66E+00	1.41E-01	1.57E-03
B-112 Tank + Leak	3.34E-01	7.74E+00	1.43E-01	1.63E-03
B-201	8.31E-01	4.53E+00	2.04E-01	5.75E-03
B-201 Leak	1.27E-04	7.12E-04	2.97E-05	8.42E-07
B-201 Tank + Leak	8.32E-01	4.53E+00	2.04E-01	5.76E-03
B-202	1.80E-01	9.94E-01	4.42E-02	1.24E-03
B-203	2.69E-01	1.46E+00	6.61E-02	1.86E-03
B-203 Leak	9.65E-05	5.43E-04	2.27E-05	6.42E-07
B-203 Tank + Leak	2.70E-01	1.47E+00	6.61E-02	1.86E-03
B-204	2.34E-01	1.27E+00	5.74E-02	1.62E-03
B-204 Leak	1.13E-06	1.14E-05	1.40E-06	7.23E-09
B-204 Tank + Leak	2.34E-01	1.27E+00	5.74E-02	1.62E-03
UPR-200-E-6	4.11E-03	2.73E-02	9.98E-04	2.49E-05
UPR-200-E-38	4.50E-01	1.13E+01	2.03E-01	2.22E-03
UPR-200-E-73	2.06E-02	2.08E-01	6.16E-03	1.34E-04
UPR-200-E-74	1.63E-06	1.28E-05	4.45E-07	1.09E-08
UPR-200-E-75	2.58E-03	1.71E-02	6.26E-04	1.56E-05

Table E-4. Intruder Dose (mrem) for All Tanks and Leaks (10 pages)

Source	Well Driller mrem	Suburban Garden mrem	Rural Pasture mrem	Commercial Farm mrem
UPR-200-E-108	5.32E-02	5.36E-01	1.59E-02	3.45E-04
UPR-200-E-109	2.40E-02	2.02E-01	6.70E-03	1.57E-04
BX-101	4.44E-01	2.46E+00	1.11E-01	3.13E-03
BX-101 Leak	6.42E-01	4.43E+00	1.71E-01	4.36E-03
BX-101 Tank + Leak	1.09E+00	6.89E+00	2.82E-01	7.49E-03
BX-102	1.46E-02	7.58E-02	3.51E-03	9.83E-05
BX-102 Leak	3.17E-01	3.38E+00	9.57E-02	1.92E-03
BX-102 Tank + Leak	3.32E-01	3.46E+00	9.92E-02	2.01E-03
BX-103	2.85E+00	1.58E+01	7.15E-01	2.01E-02
BX-104	3.85E-01	2.16E+00	9.65E-02	2.69E-03
BX-105	1.52E-01	9.19E-01	3.94E-02	1.07E-03
BX-106	3.33E-01	1.90E+00	8.45E-02	2.36E-03
BX-107	8.57E-02	5.04E-01	2.19E-02	6.05E-04
BX-108	7.34E-02	4.32E-01	1.92E-02	5.14E-04
BX-108 Leak	3.08E-01	7.68E+00	1.40E-01	1.58E-03
BX-108 Tank + Leak	3.81E-01	8.11E+00	1.59E-01	2.10E-03
BX-109	4.39E-02	2.88E-01	1.26E-02	2.99E-04
BX-110	7.46E-02	7.29E-01	2.19E-02	4.85E-04
BX-111	9.63E-02	1.25E+00	3.21E-02	5.98E-04
BX-112	2.68E-01	1.51E+00	6.75E-02	1.89E-03
BY-101	1.14E-01	1.18E+00	3.41E-02	6.90E-04
BY-102	2.25E-01	1.36E+00	5.46E-02	1.44E-03
BY-103	6.43E-02	8.91E-01	2.14E-02	3.65E-04
BY-103 Leak	2.63E-01	6.48E+00	1.18E-01	1.24E-03
BY-103 Tank + Leak	3.27E-01	7.37E+00	1.39E-01	1.60E-03
BY-104	4.89E-01	3.77E+00	1.33E-01	3.16E-03
BY-105	1.57E-01	8.64E-01	3.73E-02	9.92E-04
BY-106	4.79E-01	5.28E+00	1.42E-01	2.80E-03

Table E-4. Intruder Dose (mrem) for All Tanks and Leaks (10 pages)

Source	Well Driller mrem	Suburban Garden mrem	Rural Pasture mrem	Commercial Farm mrem
BY-107	8.89E-02	9.12E-01	2.60E-02	5.27E-04
BY-107 Leak	3.79E-01	9.35E+00	1.70E-01	1.78E-03
BY-107 Tank + Leak	4.68E-01	1.03E+01	1.96E-01	2.31E-03
BY-108	1.33E-01	1.03E+00	3.59E-02	8.44E-04
BY-108 Leak	2.63E-01	6.48E+00	1.18E-01	1.24E-03
BY-108 Tank + Leak	3.96E-01	7.52E+00	1.54E-01	2.08E-03
BY-109	1.24E-01	9.85E-01	3.32E-02	7.92E-04
BY-110	2.28E-01	2.04E+00	6.52E-02	1.46E-03
BY-111	1.60E-01	1.13E+00	4.10E-02	1.03E-03
BY-112	1.35E-01	8.45E-01	3.29E-02	8.55E-04
UPR-200-E-105	1.16E-02	7.61E-02	2.78E-03	6.93E-05
UPR-200-E-110	7.01E-03	4.59E-02	1.68E-03	4.19E-05
C-101	6.52E-01	3.68E+00	1.66E-01	4.66E-03
C-101 Leak	1.74E-01	3.01E+00	6.54E-02	9.53E-04
C-101 Tank + Leak	8.25E-01	6.69E+00	2.31E-01	5.62E-03
C-102	2.93E+00	1.65E+01	7.43E-01	2.09E-02
C-103	3.89E+00	2.19E+01	9.92E-01	2.78E-02
C-104	4.40E+00	2.46E+01	1.11E+00	3.12E-02
C-105	1.39E+00	7.96E+00	3.55E-01	9.93E-03
C-105 Leak	1.59E-01	2.94E+00	6.16E-02	8.64E-04
C-105 Tank + Leak	1.55E+00	1.09E+01	4.17E-01	1.08E-02
C-106	4.15E+00	1.94E+01	9.31E-01	2.66E-02
C-107	3.36E+00	1.74E+01	8.06E-01	2.29E-02
C-108	2.67E-02	1.51E-01	6.90E-03	1.80E-04
C-109	2.19E-01	1.38E+00	5.96E-02	1.55E-03
C-110	1.77E-01	1.04E+00	4.55E-02	1.26E-03
C-110 Leak	5.69E-02	4.66E-01	1.61E-02	3.90E-04
C-110 Tank + Leak	2.34E-01	1.50E+00	6.16E-02	1.65E-03

Table E-4. Intruder Dose (mrem) for All Tanks and Leaks (10 pages)

Source	Well Driller mrem	Suburban Garden mrem	Rural Pasture mrem	Commercial Farm mrem
C-111	5.73E-01	3.31E+00	1.53E-01	4.07E-03
C-111 Leak	1.05E+00	5.91E+00	2.66E-01	7.33E-03
C-111 Tank + Leak	1.62E+00	9.23E+00	4.19E-01	1.14E-02
C-112	5.87E-01	3.50E+00	1.54E-01	4.10E-03
C-201	1.05E+01	5.91E+01	2.66E+00	7.50E-02
C-201 Leak	1.04E+00	5.82E+00	2.62E-01	7.25E-03
C-201 Tank + Leak	1.15E+01	6.49E+01	2.92E+00	8.22E-02
C-202	8.99E+00	5.08E+01	2.29E+00	6.44E-02
C-202 Leak	9.72E-01	5.44E+00	2.44E-01	6.77E-03
C-202 Tank + Leak	9.97E+00	5.63E+01	2.53E+00	7.12E-02
C-203	1.75E+00	9.97E+00	4.46E-01	1.25E-02
C-203 Leak	9.94E-01	5.56E+00	2.50E-01	6.92E-03
C-203 Tank + Leak	2.75E+00	1.55E+01	6.96E-01	1.94E-02
C-204	1.56E-02	1.30E-01	4.22E-03	9.51E-05
C-204 Leak	9.23E-01	5.16E+00	2.32E-01	6.43E-03
C-204 Tank + Leak	9.38E-01	5.29E+00	2.36E-01	6.52E-03
UPR-200-E-81	2.84E-01	1.64E+00	7.67E-02	2.02E-03
UPR-200-E-82	3.41E-01	8.88E+00	1.59E-01	1.74E-03
UPR-200-E-86	5.15E-01	9.19E+00	1.97E-01	2.76E-03
UPR-200-E-107	7.44E-03	6.49E-02	2.16E-03	5.05E-05
S-101	2.74E-01	1.36E+00	6.11E-02	1.69E-03
S-102	1.92E+00	8.90E+00	4.02E-01	1.13E-02
S-103	2.05E+00	9.76E+00	4.37E-01	1.23E-02
S-104	2.69E-01	1.33E+00	6.00E-02	1.67E-03
S-104 Leak	2.99E-01	1.53E+00	6.96E-02	1.85E-03
S-104 Tank + Leak	5.68E-01	2.86E+00	1.30E-01	3.51E-03
S-105	7.11E-01	2.79E+00	1.04E-01	3.08E-03
S-106	3.38E-01	1.48E+00	6.14E-02	1.74E-03
S-107	5.64E-01	2.76E+00	1.25E-01	3.50E-03
S-108	9.71E-01	4.52E+00	1.99E-01	5.63E-03

Table E-4. Intruder Dose (mrem) for All Tanks and Leaks (10 pages)

Source	Well Driller mrem	Suburban Garden mrem	Rural Pasture mrem	Commercial Farm mrem
S-109	8.66E-01	3.80E+00	1.62E-01	4.62E-03
S-110	5.12E-01	2.49E+00	1.11E-01	3.11E-03
S-111	7.60E-02	4.07E-01	1.68E-02	3.93E-04
S-112	1.75E+00	7.05E+00	3.31E-01	9.55E-03
SX-101	3.70E+00	1.79E+01	8.10E-01	2.27E-02
SX-102	1.45E+00	7.04E+00	3.12E-01	8.69E-03
SX-103	6.14E-01	2.96E+00	1.33E-01	3.71E-03
SX-104	4.91E-01	2.41E+00	1.08E-01	3.02E-03
SX-104 Leak	6.63E-01	1.58E+01	2.82E-01	2.77E-03
SX-104 Tank + Leak	1.15E+00	1.82E+01	3.90E-01	5.79E-03
SX-105	4.00E+00	1.95E+01	8.77E-01	2.47E-02
SX-106	2.00E+00	9.51E+00	4.27E-01	1.21E-02
SX-107	4.10E-01	2.01E+00	9.12E-02	2.54E-03
SX-107 Leak	6.27E-01	1.22E+01	2.41E-01	2.86E-03
SX-107 Tank + Leak	1.04E+00	1.43E+01	3.32E-01	5.41E-03
SX-108	5.85E+00	2.89E+01	1.31E+00	3.64E-02
SX-108 Leak	8.32E-01	1.62E+01	3.19E-01	3.80E-03
SX-108 Tank + Leak	6.68E+00	4.51E+01	1.63E+00	4.02E-02
SX-109	2.76E-01	1.35E+00	6.09E-02	1.70E-03
SX-109 Leak	3.20E-01	6.25E+00	1.23E-01	1.46E-03
SX-109 Tank + Leak	5.96E-01	7.60E+00	1.84E-01	3.16E-03
SX-110	4.21E-01	2.07E+00	9.35E-02	2.61E-03
SX-110 Leak	2.04E-01	4.11E+00	7.77E-02	9.50E-04
SX-110 Tank + Leak	6.25E-01	6.18E+00	1.71E-01	3.56E-03
SX-111	4.41E-01	2.17E+00	9.80E-02	2.73E-03
SX-111 Leak	2.06E-01	4.36E+00	8.05E-02	9.42E-04
SX-111 Tank + Leak	6.46E-01	6.53E+00	1.78E-01	3.68E-03
SX-112	4.47E-01	2.20E+00	9.94E-02	2.77E-03
SX-112 Leak	2.54E-01	4.96E+00	9.76E-02	1.16E-03
SX-112 Tank + Leak	7.01E-01	7.16E+00	1.97E-01	3.93E-03

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Table E-4. Intruder Dose (mrem) for All Tanks and Leaks (10 pages)

Source	Well Driller mrem	Suburban Garden mrem	Rural Pasture mrem	Commercial Farm mrem
SX-113	3.91E-02	1.91E-01	8.65E-03	2.42E-04
SX-113 Leak	3.44E-01	4.01E+00	1.02E-01	1.88E-03
SX-113 Tank + Leak	3.83E-01	4.20E+00	1.10E-01	2.12E-03
SX-114	3.05E-01	1.50E+00	6.75E-02	1.88E-03
SX-115	2.89E+01	**1.41E+02**	6.42E+00	1.79E-01
SX-115 Leak	5.04E-01	5.57E+00	1.48E-01	2.77E-03
SX-115 Tank + Leak	2.94E+01	**1.47E+02**	6.57E+00	1.82E-01
T-101	7.66E-01	4.02E+00	1.79E-01	5.01E-03
T-101 Leak	1.91E-01	1.61E+00	5.25E-02	1.18E-03
T-101 Tank + Leak	9.57E-01	5.63E+00	2.31E-01	6.19E-03
T-102	6.38E-02	3.47E-01	1.50E-02	4.05E-04
T-103	2.50E-02	1.28E-01	5.65E-03	1.55E-04
T-103 Leak	3.38E-01	5.93E+00	1.23E-01	1.78E-03
T-103 Tank + Leak	3.63E-01	6.06E+00	1.29E-01	1.94E-03
T-104	2.94E-01	1.52E+00	6.85E-02	1.93E-03
T-105	4.20E-01	2.32E+00	1.00E-01	2.76E-03
T-106	1.22E-01	6.47E-01	2.86E-02	7.98E-04
T-106 Leak	1.20E+00	2.09E+01	4.35E-01	6.37E-03
T-106 Tank + Leak	1.32E+00	2.15E+01	4.63E-01	7.16E-03
T-107	5.78E-01	3.08E+00	1.36E-01	3.79E-03
T-108	1.15E-01	6.00E-01	2.68E-02	7.54E-04
T-108 Leak	7.06E-02	4.82E-01	1.78E-02	4.49E-04
T-108 Tank + Leak	1.85E-01	1.08E+00	4.46E-02	1.20E-03
T-109	6.76E-02	3.54E-01	1.58E-02	4.44E-04
T-109 Leak	2.02E-01	3.88E+00	7.69E-02	1.03E-03
T-109 Tank + Leak	2.70E-01	4.24E+00	9.27E-02	1.47E-03
T-110	2.04E-01	1.06E+00	4.75E-02	1.34E-03
T-111	4.01E-01	2.10E+00	9.37E-02	2.63E-03
T-111 Leak	2.44E-02	1.26E-01	5.69E-03	1.60E-04
T-111 Tank + Leak	4.25E-01	2.22E+00	9.94E-02	2.79E-03

Table E-4. Intruder Dose (mrem) for All Tanks and Leaks (10 pages)

Source	Well Driller mrem	Suburban Garden mrem	Rural Pasture mrem	Commercial Farm mrem
T-112	4.78E-01	2.48E+00	1.12E-01	3.15E-03
T-201	8.21E-01	4.25E+00	1.92E-01	5.40E-03
T-202	2.10E-01	1.09E+00	4.90E-02	1.38E-03
T-203	2.81E-01	1.45E+00	6.55E-02	1.84E-03
T-204	2.31E-01	1.20E+00	5.39E-02	1.52E-03
TX-101	2.23E+00	1.09E+01	4.94E-01	1.38E-02
TX-102	1.10E+00	6.09E+00	2.47E-01	6.65E-03
TX-103	1.17E+00	6.46E+00	2.63E-01	7.08E-03
TX-104	9.35E-01	4.80E+00	2.10E-01	5.74E-03
TX-105	1.08E+00	5.96E+00	2.42E-01	6.52E-03
TX-106	1.75E+00	8.29E+00	3.75E-01	1.06E-02
TX-107	1.72E+01	8.40E+01	3.77E+00	1.06E-01
TX-107 Leak	6.18E-01	1.31E+01	2.45E-01	2.70E-03
TX-107 Tank + Leak	1.78E+01	9.70E+01	4.01E+00	1.09E-01
TX-108	1.00E+00	5.57E+00	2.26E-01	6.08E-03
TX-109	3.96E-01	1.98E+00	8.81E-02	2.47E-03
TX-110	1.06E+00	5.84E+00	2.37E-01	6.40E-03
TX-111	9.96E-01	5.51E+00	2.24E-01	6.05E-03
TX-112	1.17E+00	6.47E+00	2.63E-01	7.10E-03
TX-113	1.72E-01	1.85E+00	4.69E-02	9.22E-04
TX-114	7.25E-01	4.00E+00	1.63E-01	4.39E-03
TX-115	1.18E+00	6.52E+00	2.65E-01	7.14E-03
TX-116	4.14E-01	2.32E+00	9.29E-02	2.49E-03
TX-117	6.06E-01	3.31E+00	1.36E-01	3.68E-03
TX-118	3.04E+01	**1.48E+02**	6.67E+00	1.88E-01
UPR-200-W-100	6.16E-03	3.69E-02	1.35E-03	3.36E-05
UPR-200-W-12	1.33E-02	8.50E-02	3.16E-03	8.02E-05
TY-101	1.21E-01	6.10E-01	2.72E-02	7.64E-04
TY-101 Leak	2.58E-01	5.37E+00	1.02E-01	1.16E-03
TY-101 Tank + Leak	3.79E-01	5.98E+00	1.29E-01	1.92E-03

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Table E-4. Intruder Dose (mrem) for All Tanks and Leaks (10 pages)

Source	Well Driller mrem	Suburban Garden mrem	Rural Pasture mrem	Commercial Farm mrem
TY-102	3.55E-02	1.33E-01	5.56E-03	1.61E-04
TY-103	1.85E-01	9.68E-01	4.22E-02	1.16E-03
TY-103 Leak	4.42E-01	9.47E+00	1.77E-01	1.96E-03
TY-103 Tank + Leak	6.27E-01	1.04E+01	2.19E-01	3.11E-03
TY-104	2.87E-01	1.48E+00	6.52E-02	1.80E-03
TY-104 Leak	1.66E-06	8.59E-06	3.74E-07	1.04E-08
TY-104 Tank + Leak	2.87E-01	1.48E+00	6.52E-02	1.80E-03
TY-105	5.58E-02	3.09E-01	1.39E-02	3.49E-04
TY-105 Leak	1.61E-01	1.24E+00	4.12E-02	9.67E-04
TY-105 Tank + Leak	2.17E-01	1.55E+00	5.51E-02	1.32E-03
TY-106	1.75E-02	9.68E-02	4.20E-03	1.08E-04
TY-106 Leak	1.34E-01	1.03E+00	3.42E-02	8.03E-04
TY-106 Tank + Leak	1.51E-01	1.13E+00	3.84E-02	9.11E-04
U-101	1.60E-01	8.43E-01	3.92E-02	9.81E-04
U-101 Leak	2.20E-01	2.56E+00	6.51E-02	1.20E-03
U-101 Tank + Leak	3.80E-01	3.41E+00	1.04E-01	2.18E-03
U-102	1.28E+00	7.05E+00	2.90E-01	7.79E-03
U-103	1.24E+00	5.96E+00	2.62E-01	7.32E-03
U-104	5.22E-01	2.57E+00	1.16E-01	3.24E-03
U-104 Leak	3.41E-01	3.10E+00	9.20E-02	2.00E-03
U-104 Tank + Leak	8.63E-01	5.67E+00	2.08E-01	5.23E-03
U-105	2.26E+00	1.11E+01	4.94E-01	1.39E-02
U-106	6.38E+00	3.06E+01	1.39E+00	3.91E-02
U-107	2.03E+00	1.13E+01	4.58E-01	1.24E-02
U-108	4.56E-01	2.92E+00	1.07E-01	2.73E-03
U-109	2.77E-01	2.85E+00	7.36E-02	1.47E-03
U-110	2.47E-01	1.23E+00	5.51E-02	1.53E-03
U-110 Leak	7.70E-02	1.37E+00	2.77E-02	3.66E-04
U-110 Tank + Leak	3.24E-01	2.59E+00	8.29E-02	1.90E-03

Table E-4. Intruder Dose (mrem) for All Tanks and Leaks (10 pages)

Source	Well Driller mrem	Suburban Garden mrem	Rural Pasture mrem	Commercial Farm mrem
U-111	1.40E+00	8.20E+00	3.20E-01	8.45E-03
U-112	3.40E-02	1.81E-01	8.41E-03	2.09E-04
U-112 Leak	2.27E-01	2.51E+00	6.57E-02	1.25E-03
U-112 Tank + Leak	2.61E-01	2.70E+00	7.41E-02	1.46E-03
U-201	4.29E-03	3.47E-02	1.14E-03	2.63E-05
U-202	5.35E-03	3.87E-02	1.36E-03	3.30E-05
U-203	3.61E-03	2.98E-02	9.64E-04	2.21E-05
U-204	1.11E-01	5.51E-01	2.46E-02	6.90E-04
UPR-200-W-132	8.07E-02	7.33E-01	2.17E-02	4.72E-04
UPR-200-W-24	1.97E-02	1.83E-01	5.39E-03	1.16E-04

The dose from the soil plume alone is listed as the tank number followed by the word "Leak."

The dose from a well that passes through both the tank residual and tank leak plume is shown as the tank number followed by the words "Tank + Leak."

Bold text indicates that the calculated dose exceeds the performance objective for the scenario (500 mrem/yr for well driller and 100 mrem/yr for post-intruder resident).

1 **E3.0 AIR PATHWAY RESULTS FOR ALL WASTE ITEMS**

2 Table E-5 shows the projected maximum radon-222 flux at the ground surface for the tank
3 residuals and releases to soil in each waste management area. Several tanks have experienced
4 significant leaks into the soil. The radon source could be the tank, the leak plume, or both.
5 Radon fluxes at the ground surface for all three combinations are listed in the table. The flux
6 from the tank alone is listed as the tank number without adjectives. The flux from the soil plume
7 alone is listed as the tank number followed by the word "Leak." The flux from both the tank
8 residual and tank leak plume is shown as the tank number followed by the words "Tank + Leak."

Table E-5. Eventual Maximum Radon Flux (6 pages)

Tank	Flux Ci/m^2/s	Time years	Tank	Flux Ci/m^2/s	Time years
200 East Area Sources					
A Tank Farm					
A-101	4.21E-01	508,103	A-104 Leak	3.06E-03	859,231
A-102	2.73E+00	505,210	A-104 Tank + Leak	8.25E-02	686,291
A-103	2.45E-01	508,605	A-105	2.54E-03	194,349
A-103 Leak	3.38E-03	500,825	A-105 Leak	4.23E-03	512,998
A-103 Tank + Leak	2.48E-01	508,499	A-105 Tank + Leak	6.78E-03	393,459
A-104	7.94E-02	679,621	A-106	3.06E-02	480,615
AX Tank Farm					
AX-101	3.85E-01	538,316	AX-102 Tank + Leak	3.06E-01	448,667
AX-102	3.02E-01	449,576	AX-103	8.46E-02	303,636
AX-102 Leak	3.44E-03	368,871	AX-104	4.46E-02	507,638
B Tank Farm					
B-101	8.83E-01	688,984	B-201	5.44E-03	682,070
B-102	1.80E-01	2,226,608	B-201 Leak	2.59E-06	1,111,393
B-103	6.02E-01	2,206,871	B-201 Tank + Leak	5.44E-03	682,274
B-104	2.88E-01	2,220,524	B-202	1.20E-02	375,156
B-105	2.68E-01	2,226,371	B-203	2.91E-04	407,443
B-106	9.91E-01	2,212,718	B-203 Leak	1.76E-06	2,021,405
B-107	1.23E-01	2,190,235	B-203 Tank + Leak	2.92E-04	417,150
B-107 Leak	3.14E-03	558,241	B-204	3.60E-05	187,205
B-107 Tank + Leak	1.26E-01	2,149,568	B-204 Leak	1.65E-06	750,901
B-108	1.20E-01	890,561	B-204 Tank + Leak	3.76E-05	211,914
B-109	5.07E-01	535,206	UPR-200-E-6	2.94E-03	2,320,297

Table E-5. Eventual Maximum Radon Flux (6 pages)

Tank	Flux Ci/m²/s	Time years	Tank	Flux Ci/m²/s	Time years
B-110	1.80E-02	1,103,525	UPR-200-E-38	3.33E-03	557,219
B-110 Leak	3.30E-04	473,240	UPR-200-E-73	3.32E-02	2,225,993
B-110 Tank + Leak	1.83E-02	1,092,149	UPR-200-E-74	2.35E-06	2,235,620
B-111	5.03E-03	558,766	UPR-200-E-75	1.56E-03	2,300,176
B-112	2.00E-02	515,070	UPR-200-E-108	7.41E-02	2,302,013
B-112 Leak	2.88E-03	548,040	UPR-200-E-109	2.32E-03	2,204,963
B-112 Tank + Leak	2.29E-02	519,220			
BX Tank Farm					
BX-101	1.49E-01	655,358	BX-106	1.33E-01	2,249,577
BX-101 Leak	3.17E-03	551,571	BX-107	8.74E-02	2,227,741
BX-101 Tank + Leak	1.52E-01	653,202	BX-108	5.81E-01	2,212,351
BX-102	3.71E-02	513,223	BX-108 Leak	3.10E-03	552,942
BX-102 Leak	3.43E+00	2,177,070	BX-108 Tank + Leak	5.84E-01	2,203,550
BX-102 Tank + Leak	3.47E+00	2,159,270	BX-109	1.54E+00	2,211,162
BX-103	1.99E-01	670,396	BX-110	8.93E-02	646,088
BX-104	1.23E+00	801,562	BX-111	4.18E-02	1,710,039
BX-105	1.29E+00	2,210,517	BX-112	5.80E-02	2,216,582
BY Tank Farm					
BY-101	1.40E-01	2,152,970	BY-107 Tank + Leak	3.51E-01	661,124
BY-102	5.77E-02	414,847	BY-108	8.62E-01	1,982,716
BY-103	5.57E-02	425,109	BY-108 Leak	2.82E-03	413,489
BY-103 Leak	2.82E-03	413,013	BY-108 Tank + Leak	8.65E-01	1,977,600
BY-103 Tank + Leak	5.85E-02	424,526	BY-109	3.08E-02	419,805
BY-104	9.85E-01	807,418	BY-110	7.18E-01	749,138
BY-105	2.58E-01	865,512	BY-111	3.38E-02	414,611
BY-106	1.47E+00	1,494,093	BY-112	2.16E-01	749,677
BY-107	3.48E-01	663,367	UPR-200-E-105	3.52E-03	2,267,396
BY-107 Leak	3.15E-03	412,908	UPR-200-E-110	3.25E-03	2,238,099

Table E-5. Eventual Maximum Radon Flux (6 pages)

Tank	Flux Ci/m²/s	Time years	Tank	Flux Ci/m²/s	Time years
*C Tank Farm*					
C-101	4.31E-01	2,279,914	C-111 Tank + Leak	3.20E-01	2,212,435
C-101 Leak	2.85E-03	2,009,442	C-112	1.49E+00	2,231,861
C-101 Tank + Leak	4.33E-01	2,278,136	C-201	5.27E+00	2,323,034
C-102	3.45E-01	447,622	C-201 Leak	2.57E-03	958,497
C-103	1.01E-01	1,219,885	C-201 Tank + Leak	5.27E+00	2,322,369
C-104	7.35E-01	288,160	C-202	4.64E+00	610,051
C-105	1.65E-01	2,252,662	C-202 Leak	2.51E-03	937,591
C-105 Leak	2.53E-03	456,685	C-202 Tank + Leak	4.64E+00	610,228
C-105 Tank + Leak	1.67E-01	2,225,453	C-203	6.84E+00	2,482,765
C-106	9.57E-03	258,890	C-203 Leak	2.63E-03	961,701
C-107	1.52E-01	794,413	C-203 Tank + Leak	6.85E+00	2,482,180
C-108	1.48E-02	2,219,349	C-204	2.76E+00	3,472,304
C-109	3.30E-01	606,882	C-204 Leak	2.52E-03	937,161
C-110	1.36E-01	2,227,184	C-204 Tank + Leak	2.77E+00	3,469,998
C-110 Leak	3.06E-03	2,209,808	UPR-200-E-81	3.68E-03	501,139
C-110 Tank + Leak	1.39E-01	2,226,800	UPR-200-E-82	3.17E-03	556,612
C-111	3.17E-01	2,214,523	UPR-200-E-86	3.38E-03	863,520
C-111 Leak	3.06E-03	1,996,105	UPR-200-E-107	1.24E-03	2,220,072
*200 West Area Sources*					
*S Tank Farm*					
S-101	1.64E-01	1,772,222	S-106	7.93E-02	585,461
S-102	1.57E+00	1,605,068	S-107	1.60E-01	514,909
S-103	2.56E-01	677,989	S-108	1.90E-01	607,194
S-104	8.95E-02	2,004,465	S-109	1.41E-01	1,046,349
S-104 Leak	4.12E-03	338,249	S-110	1.75E-01	1,778,560
S-104 Tank + Leak	9.36E-02	1,931,145	S-111	1.62E-02	631,053
S-105	2.52E-01	616,053	S-112	5.32E-01	620,678

Table E-5. Eventual Maximum Radon Flux (6 pages)

Tank	Flux Ci/m²/s	Time years	Tank	Flux Ci/m²/s	Time years
SX Tank Farm					
SX-101	5.73E-01	574,410	SX-110	8.09E-02	2,127,913
SX-102	1.64E-01	620,181	SX-110 Leak	2.18E-03	541,740
SX-103	4.58E-02	646,166	SX-110 Tank + Leak	8.31E-02	2,086,405
SX-104	9.40E-02	1,890,186	SX-111	8.78E-02	2,225,656
SX-104 Leak	3.18E-03	556,510	SX-111 Leak	2.59E-03	524,397
SX-104 Tank + Leak	9.72E-02	1,846,565	SX-111 Tank + Leak	9.04E-02	2,176,943
SX-105	1.48E-01	756,308	SX-112	8.98E-02	2,247,599
SX-106	1.17E-01	553,724	SX-112 Leak	3.03E-03	444,327
SX-107	8.27E-02	2,273,852	SX-112 Tank + Leak	9.29E-02	2,188,775
SX-107 Leak	3.62E-03	444,863	SX-113	2.16E-03	2,208,084
SX-107 Tank + Leak	8.63E-02	2,197,073	SX-113 Leak	3.43E-03	2,214,145
SX-108	3.34E-01	2,206,465	SX-113 Tank + Leak	5.59E-03	2,211,801
SX-108 Leak	3.75E-03	444,332	SX-114	5.47E-02	1,612,462
SX-108 Tank + Leak	3.37E-01	2,186,908	SX-115	2.54E-01	305,983
SX-109	4.71E-02	999,708	SX-115 Leak	4.05E-03	361,670
SX-109 Leak	3.21E-03	445,174	SX-115 Tank + Leak	2.58E-01	306,857
SX-109 Tank + Leak	5.04E-02	964,331			
T Tank Farm					
T-101	4.24E-01	342,987	T-108 Leak	2.90E-03	2,210,594
T-101 Leak	2.93E-03	421,059	T-108 Tank + Leak	3.58E-02	2,235,810
T-101 Tank + Leak	4.27E-01	343,522	T-109	9.70E-02	2,281,968
T-102	5.07E-01	2,248,727	T-109 Leak	2.44E-03	579,966
T-103	3.12E-01	1,081,966	T-109 Tank + Leak	9.95E-02	2,240,271
T-103 Leak	2.73E-03	521,556	T-110	2.18E-03	1,255,194
T-103 Tank + Leak	3.14E-01	1,077,100	T-111	2.83E-01	2,261,760
T-104	6.38E-02	359,170	T-111 Leak	2.97E-04	2,135,353
T-105	8.53E-02	560,151	T-111 Tank + Leak	2.83E-01	2,261,628

Table E-5. Eventual Maximum Radon Flux (6 pages)

Tank	Flux Ci/m²/s	Time years	Tank	Flux Ci/m²/s	Time years
T-106	1.32E-01	391,595	T-112	3.85E-02	2,171,122
T-106 Leak	3.33E-03	520,241	T-201	5.17E-05	187,143
T-106 Tank + Leak	1.35E-01	394,759	T-202	3.52E-03	1,978,207
T-107	3.82E-01	2,104,303	T-203	1.36E-04	307,102
T-108	3.29E-02	2,238,028	T-204	6.59E-05	245,866
TX Tank Farm					
TX-101	9.14E-01	2,233,569	TX-110	1.58E-01	541,282
TX-102	7.23E-01	1,045,676	TX-111	1.58E-01	551,980
TX-103	1.58E-01	520,235	TX-112	1.61E-01	522,622
TX-104	1.14E-01	548,062	TX-113	8.30E-02	692,341
TX-105	8.84E-01	1,370,713	TX-114	1.05E-01	534,740
TX-106	5.05E-01	686,205	TX-115	1.87E-01	543,456
TX-107	1.76E-01	450,936	TX-116	9.62E-02	689,495
TX-107 Leak	3.41E-03	514,928	TX-117	1.15E-01	612,994
TX-107 Tank + Leak	1.80E-01	452,151	TX-118	1.75E-01	420,618
TX-108	1.07E+00	1,777,996	UPR-200-W-100	3.17E-03	2,218,360
TX-109	2.27E-01	2,226,024	UPR-200-W-12	1.24E-03	2,219,277
TY Tank Farm					
TY-101	4.42E-02	2,224,443	TY-104 Leak	4.64E-06	740,956
TY-101 Leak	2.85E-03	509,917	TY-104 Tank + Leak	8.09E-01	2,207,132
TY-101 Tank + Leak	4.70E-02	2,120,540	TY-105	3.40E-01	2,210,796
TY-102	3.09E-02	646,658	TY-105 Leak	3.57E-03	2,204,955
TY-103	8.96E-01	826,596	TY-105 Tank + Leak	3.43E-01	2,210,735
TY-103 Leak	3.23E-03	512,622	TY-106	1.06E-01	2,210,869
TY-103 Tank + Leak	8.99E-01	825,468	TY-106 Leak	3.50E-03	2,209,627
TY-104	8.09E-01	2,207,140	TY-106 Tank + Leak	1.10E-01	2,210,830

Table E-5. Eventual Maximum Radon Flux (6 pages)

Tank	Flux Ci/m^2/s	Time years	Tank	Flux Ci/m^2/s	Time years
U Tank Farm					
U-101	3.76E-01	2,347,055	U-110	4.19E-01	2,192,308
U-101 Leak	2.99E-03	2,227,987	U-110 Leak	6.71E-04	580,652
U-101 Tank + Leak	3.79E-01	2,346,115	U-110 Tank + Leak	4.19E-01	2,189,729
U-102	1.82E-01	864,058	U-111	1.80E-01	770,173
U-103	1.05E-01	632,497	U-112	1.27E-01	2,147,602
U-104	1.07E-01	2,308,717	U-112 Leak	5.29E-03	2,195,677
U-104 Leak	1.01E-01	2,236,178	U-112 Tank + Leak	1.33E-01	2,149,517
U-104 Tank + Leak	2.08E-01	2,273,639	U-201	5.89E-04	700,337
U-105	7.66E-01	687,622	U-202	5.78E-04	696,803
U-106	2.03E-01	555,238	U-203	2.93E-04	690,518
U-107	7.15E-02	556,462	U-204	4.11E-02	739,258
U-108	2.08E-01	347,280	UPR-200-W-132	8.27E-02	2,254,935
U-109	3.72E-02	605,845	UPR-200-W-24	5.74E-02	2,283,224

The flux from the tank alone is listed as the tank number without adjectives.

The flux from the soil plume alone is listed as the tank number followed by the word "Leak."

The flux from both the tank residual and tank leak plume is shown as the tank number followed by the words "Tank + Leak."

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APPENDIX F

QUALITY ASSURANCE

CONTENTS

TERMS

ASME	American Society of Mechanical Engineers
CFR	*Code of Federal Regulations*
CH2M HILL	CH2M HILL Hanford Group, Inc.
CIO	Chief Information Officer
COTS	commercial off-the-shelf software
DMT	Decision Management Tool
DOE	U.S. Department of Energy
HTWOS	Hanford Tank Waste Operations Simulator
SQAP	software quality assurance plan
SST PA	*Initial Single-Shell Tank System Performance Assessment for the Hanford Site*
STOMP	Subsurface Transport Over Multiple Phases

April 2006

F1.0   DOCUMENT PRODUCTION QUALITY ASSURANCE

The production of *Initial Single-Shell Tank System Performance Assessment for the Hanford Site* (SST PA) was governed by the CH2M HILL Hanford Group, Inc. (CH2M HILL) quality assurance program, which is described in TFC-PLN-02, "Quality Assurance Program Description." The CH2M HILL quality assurance program implements the requirements of 10 *Code of Federal Regulations* (CFR) 830, Subpart A and U.S. Department of Energy (DOE) Order 414.1B, and is applied on a graded basis to all CH2M HILL activities and products.

Detailed implementation of TFC-PLN-02 is found in "Graded Quality Assurance" (TFC-ESHQ-Q_ADM-C-01) and "Document Control" (TFC-BSM-IRM_DC-C-01). TFC-ESHQ-Q_ADM-C-01 defines the graded approach applied to CH2M HILL products. TFC-BSM-IRM_DC-C-01 has detailed flow charts for determining levels of control and quality assurance based on document purpose, use, and content. SST PA production activities followed these procedures.

Several more detailed procedures guide specific SST PA activities including software and model development, laboratory investigations, and document consistency and accuracy. These activities and the procedures guiding them are described in the following sections.

F1.1   SOFTWARE AND MODEL DEVELOPMENT

A number of models and software packages used in the SST PA to estimate groundwater impacts from contamination were either developed or purchased by CH2M HILL. "Software Life Cycle Standard" (TFC-BSM-IRM-STD-01), based on American Society of Mechanical Engineers (ASME) NQA-1 1994 (Subpart 2.7), describes how to define the level of quality assurance applied to developed or purchased software. TFC-BSM-IRM-STD-01 contains a checklist for determining if a software application is quality affecting. If quality affecting, the software must have a software quality assurance plan (SQAP) approved by the CH2M HILL Chief Information Officer (CIO) and the CH2M HILL quality assurance group prior to application, purchase, or development.

F1.1.1   Decision Management Tool

Decision Management Tool©[1] (DMT) was developed by CH2M HILL to calculate estimated groundwater concentrations of selected constituents based on user-defined tank closure scenarios, predict the resulting human health risk, and compare predicted risk to regulatory criteria. DMT was developed under the procedure, "Custom Software Development, Implementation, and Management" (TFC-BSM-IRM_HS-C-03).

TFC-BSM-IRM_HS-C-03 requires custom software development to adhere to the SQAP developed under TFC-BSM-IRM_HS-C-01. TFC-BSM-IRM_HS-C-03 also requires the software owner to identify software test requirements using criteria provided in the procedure, write a test plan based on those criteria, and submit the test plan to the CH2M HILL CIO for approval. The approved test plan is used to test the finished custom software and the results are documented in a test report submitted to the CH2M HILL CIO for approval prior to using the tested software.

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[1] Copyright pending.

1 Using the criteria set forth in TFC-BSM-IRM_HS-C-01, DMT was determined to be quality
2 affecting. Accordingly, as required in TFC-BSM-IRM_HS-C-03, it has a specific configuration
3 management plan (Watson 2005f), approved quality assurance plan (Watson 2005d), approved
4 software test plan (Watson 2005e), and documented and approved test results (Watson 2005c).

5 **F1.1.2   Hanford Tank Waste Operations Simulator**

6 Hanford Tank Waste Operations Simulator (HTWOS) is a computerized dynamic flowsheet
7 mass balance model predicting tank waste mass and activity resulting from retrieval and transfer
8 activities. The inventories predicted in HTWOS for single-shell tanks are used as DMT inputs.
9 HTWOS is adapted from G2[2], a commercially available process control engine. Because G2 is
10 commercially available, HTWOS was developed under both TFC-BSM-IRM_HS-C-03,
11 "Custom Software Development, Implementation and Management," and
12 TFC-BSM-IRM_HS-C-02, "COTS Software Acquisition, Implementation and Management."

13 TFC-BSM-IRM_HS-C-02 requires software to be purchased using the procedure "Procurement
14 of Items (Materials)" (TFC-BSM-CP_CPR-C-06) in order to ensure that the requirements of
15 TFC-PLN-02 are met.

16 TFC-BSM-IRM_HS-C-02 also requires developing a test plan for commercial off-the-shelf
17 (COTS) software. Both the test plan and the accompanying test results document must be
18 approved by the CH2M HILL CIO prior to software use.

19 Using the criteria set forth in TFC-BSM-IRM_HS-C-01, HTWOS was determined to be
20 quality affecting. Accordingly, as required in TFC-BSM-IRM_HS-C-03, it has a specific
21 configuration management plan (Naiknimbalkar 2005d), approved quality assurance plan
22 (Naiknimbalkar 2005c), and an approved software test plan and test results document
23 (Naiknimbalkar 2005a).

24 **F1.1.3   Subsurface Transport Over Multiple Phases**

25 Subsurface Transport Over Multiple Phases (STOMP)[3] was chosen to model contaminant
26 transport through the vadose zone and groundwater out to the waste management area fenceline.
27 STOMP meets the requirements of *Computer Code Selection Criteria for Flow and Transport*
28 *Code(s) To Be Used in Vadose Zone Calculations for Environmental Analyses in the Hanford*
29 *Site's Central Plateau* (Mann et al. 1999), and has been used for a number of risk assessments on
30 the Hanford Site (e.g., Knepp 2002a, 2002b). STOMP was purchased in accordance with the
31 requirements in TFC-BSM-IRM_HS-C-02.

32 Using the criteria in TFC-BSM-IRM_HS-C-01, STOMP was determined to be quality affecting.
33 Accordingly, as required in TFC-BSM-IRM_HS-C-02, STOMP has a specific configuration
34 management plan (McMahon 2005d), an approved quality assurance plan (McMahon 2005c),
35 an approved software test plan (McMahon 2005b), and an approved test results document
36 (McMahon 2005a).

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[2] G2 is a trademark of the Gensym Corporation, Burlington, Massachusetts.
[3] Subsurface Transport Over Multiple Phases (STOMP) is copyrighted by Battelle Memorial Institute, 1996.

1 **F1.2   LABORATORY INVESTIGATIONS**

2 Laboratory investigations associated with the analysis of past tank leak soil and contaminant data
3 supporting the SST PA were conducted by Pacific Northwest National Laboratory (PNNL).
4 PNNL laboratory work in support of the SST PA complies with *Hanford Analytical Services*
5 *Quality Assurance Requirements Documents* (DOE-RL 1998). The PNNL quality assurance
6 program is based on the requirements of DOE O 414.1B and 10 CFR 830, Subpart A.

7 **F1.3   DOCUMENT CONSISTENCY AND ACCURACY**

8 Each SST PA chapter was peer reviewed for internal consistency and technical accuracy by other
9 SST PA authors. The first three SST PA chapters were publicly released (DOE-ORP 2005) and
10 submitted for review by the following Department of Energy (DOE) organizations:

11 • DOE-Headquarters
12 • Idaho National Engineering and Environmental Laboratory
13 • Savannah River Site.

14 The initial draft of the complete SST PA was reviewed by subject matter experts outside the
15 SST PA contributor team. These subject matter experts come from other Hanford Site
16 contractors and have authored similar documents (i.e., performance assessments and risk
17 assessments). These experts reviewed the SST PA for robustness of the analyses, internal
18 consistency, and technical accuracy. Since many of the reviewers had authored documents
19 referenced by the SST PA, they were qualified to look for agreement between the SST PA and
20 their previously published documents. These experts also highlighted differences in assumptions
21 or data that could lead to differing predictions.

22 After addressing comments by both Hanford Site and external subject matter experts, an internal
23 CH2M HILL review of Predecisional Draft, Revision B, of the SST PA was completed.
24 The resulting Predecisional Draft, Revision C, was transmitted to DOE-Headquarters and
25 DOE, Office of River Protection (ORP) for review and comment.

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APPENDIX G
PREPARERS

1   **CONTENTS**

5

G1.0   PREPARERS

G1.1   OFFICE OF RIVER PROTECTION

Robert W. Lober, Physical Scientist

> B.S.   Soil Science (Soil Physics), Colorado State University
> M.S.   Soil Science, (Soil Chemistry and Parametric Statistics),
> Colorado State University

Mr. Lober has 17 years experience as a Research Soil Scientist with the USDA Agricultural Research Service and 10 years as a Physical Scientist with the Department of Energy, Office of River Protection. Mr. Lober's areas of professional expertise include disturbed ecosystem function, decision support tools, and RCRA-NEPA environmental compliance. Mr. Lober is a Certified Professional Soil Scientist (ARCPACS). His work with the USDA included evaluation of reclamation strategies on disturbed lands, developing decision support tools optimizing various management systems, assessing terrestrial inputs for Global Climate Change Models, and elucidating nutrient cycling processes in disturbed ecosystems. As a physical scientist for DOE, Mr. Lober has been the Environmental Compliance lead for RCRA and the technical representative for two environmental impact statements (tank waste remediation EIS, safe interim storage EIS) and numerous environmental assessments. His project experience includes serving as Tank Farm Closure Project Manager and he is currently Program Manager for Closure, Retrieval Risk Assessments for the Office of River Protection and a member of the Hanford Risk Assessment Configuration Management Group, integrating and managing risk assessments for the Hanford Site.

Roger A. Quintero, General Engineer

> B.S.   Chemical Engineering, Rice University
> M.S.   Natural Resources, University of Texas at San Antonio

Mr. Quintero has 10 years of experience of oversight of Department of Energy waste management facilities at the Hanford Site. He currently is the Program Manager for the Single-Shell Tank Retrieval and Closure Project with responsibility for the performance assessment for waste management area C at the Hanford Site released in 2003, the SST RCRA closure plan and risk assessment released in 2004, and this SST performance assessment. Previously, he was the Program Manager and DOE Facility Representative for the Liquid Waste Processing Facilities. Prior to working for DOE, Mr. Quintero worked for 6 years as an engineer in private industry, primarily with design and operation of industrial waste treatment systems, and obtained certification as a registered professional engineer in the state of Texas.

1 **Robert M. Yasek**, Physical Scientist

      B.S.   Geophysics, New Mexico Institute of Mining and Technology

2 Mr. Yasek has over 19 years of experience in leadership and management of government
3 projects, both military and civilian. Prior to working at the Hanford Site, he was the Department
4 of Energy Lead for Thermal Testing for the DOE Yucca Mountain Project. Additional duties at
5 the Yucca Mountain Project included project management of borehole geophysics for the
6 high-level waste repository program. Mr. Yasek's military experience includes project
7 management for flight testing of advanced weapons systems for the U.S. Air Force and
8 operations of radar and command, control and communications (C3) systems. Between his
9 military and civilian government service, Mr. Yasek worked as a field geophysicist for a
10 privately owned company, specializing in borehole geophysics.

11 **G1.2   CORE TEAM**

12 **Frank J. Anderson**, Task Leader, Tank Farm Vadose Zone Project, CH2M HILL Hanford
13 Group, Inc.

      B.S.   Geological Engineering, Colorado School of Mines
      M.S.   Geological Engineering, University of Arizona

14 Mr. Anderson has over 35 years of experience as a geological engineer, environmental
15 consultant, government manager, and professor involving environmental characterization,
16 compliance and remediation, mining, geology, water resources development, and program and
17 project management. He has worked as a consultant at five Department of Energy sites during
18 the past decade: Hanford and Oak Ridge reservations, Portsmouth and Paducah Gaseous
19 Diffusion Plants, and Idaho National Engineering and Environmental Laboratory. He has also
20 been a manager for the U.S. Geological Survey and the U.S. Office of Surface Mining, and an
21 assistant professor of geology. Mr. Anderson was responsible for fiscal year 2002 interim
22 measures engineering design and construction activities for the Tank Farm Vadose Zone Project.
23 Mr. Anderson has been the manager of Tank Farm Vadose Zone Project/Risk Assessments since
24 July 2003. His responsibilities include this performance assessment.

25 **Michael P. Connelly**, Scientist, CH2M HILL Hanford Group, Inc.

      B.S.   Geology, University of Utah
      M.S.   Geology, University of Utah

26 Mr. Connelly led the risk assessment team for the Tank Farm Vadose Zone Project and is a
27 co-author of this document. He has over 16 years of experience in environmental geohydrology
28 including project management, groundwater modeling, and using computer techniques to analyze
29 and interpret field data for remedial action and site characterization activities.

1 **Dwayne Crumpler**, Senior Hydrogeologist, Columbia Engineering and Environmental
2 Services, Inc.

  B.S.   Geology, Lamar University
  M.S.   Geology, Baylor University

3 Mr. Crumpler has over 15 years of experience in groundwater field investigations related to
4 RCRA facility investigations and CERCLA remedial investigations at municipal landfills, and
5 Department of Defense and Department of Energy facilities. He serves as a Senior Geologist
6 and Regulatory Specialist for the preparation of various RCRA and NEPA documents related to
7 the Hanford Site. He has conducted and analyzed seismic field studies, aquifer pumping tests
8 and slug tests, installed monitoring wells and soil borings, and conducted groundwater and
9 surface-water sampling programs at DOE and DOD facilities. He has prepared the site-specific
10 work plans associated with the single-shell tanks RCRA Corrective Action program and has been
11 involved in the SST retrieval program. He was the coordinator for the preparation of the human
12 health risk analyses, regulatory analyses, introduction, approach, conclusions, and
13 recommendations for this document.

14 **R. Douglas Evans**, Geologist, Columbia Energy and Environmental Services, Inc.

  B.S.   Geology, University of Illinois
  M.S.   Geology, University of Idaho

15 Mr. Evans has over 15 years experience conducting human health risk assessments and
16 hydrogeologic investigations in support of federal agency RCRA, CERCLA, and NEPA
17 compliance programs. His focus over the past 10 years has been on human health risk analysis
18 to support Department of Energy decisions on Hanford Site radioactive tank waste retrieval, tank
19 farm corrective actions, and tank farm closure. He has led the development of pre-retrieval
20 human health risk information for a series of single-shell tank waste retrieval work plans. He has
21 also led the development of human health risk information for a series of field investigation
22 reports documenting the nature and extent of contamination related to past waste releases at SST
23 waste management areas. For this performance assessment, he contributed to the human health
24 risk assessment and to the compilation of data and information for Chapters 2.0, 4.0, and 6.0.

1 **Vicky L. Freedman**, Senior Research Scientist, Hydrology Group, Pacific Northwest National
2 Laboratory

> B.A   English Language and Literature, University of Michigan
> M.S   Watershed Management (Watershed Hydrology), University of
>   Arizona
> Ph.D Environmental Sciences, Ohio State University

3 Dr. Freedman's research activities have included theoretical and numerical studies of coupled
4 hydrodynamics, contaminant transport, and geochemistry in environmental systems. Over the
5 past few years, she has been involved in both forward prediction and inverse modeling of tank
6 farm wastes at the Hanford Site, and was a major contributor to the vadose zone modeling for the
7 *Field Investigation Report for Waste Management Area B-BX-BY*. In support of the Hanford
8 Low-Activity Waste Glass Project, she has investigated methods for identifying unsaturated
9 hydraulic properties of the fractured glass. For this project, she served as project manager of the
10 vadose zone modeling and groundwater pathway effort for Pacific Northwest National
11 Laboratory.

12 **Thomas E. Jones**, Chemist, CH2M HILL Hanford Group, Inc.

> B.S.   Chemistry, College of Great Falls
> Ph.D. Chemistry, Washington State University

13 Dr. Jones has over 20 years experience at the Hanford Site in the areas of tank waste
14 characterization, development of tank waste inventory estimates, and tank farm vadose zone
15 investigations. Over the past 4 years, Dr. Jones has led the task developing tank leak inventory
16 estimates. He supplied the process chemistry and facility history sections in Appendix B and the
17 inventory sections in Chapter 3.0.

18 **Raziuddin Khaleel**, Consulting Engineer, Nuclear and Environmental Initiatives, Fluor Federal
19 Services

> B.S.   Civil Engineering, Bangladesh University of Engineering and
>   Technology
> M.S.   Water Science and Engineering, Asian University of
>   Technology
> Ph.D. Soil and Water Engineering, Texas A&M University

20 Dr. Khaleel has over 25 years of experience in groundwater hydrology and numerical
21 simulations of subsurface flow and transport. He was a key contributor to the Hanford Site
22 *Performance Assessment for the Disposal of Low-Level Waste in the 200 West Area Burial*
23 *Grounds,* the *Performance Assessment for the Disposal of Low-Level Waste in the 200 East Area*
24 *Burial Grounds*, and the *Hanford Immobilized Low-Activity Tank Waste Performance*
25 *Assessment*, particularly in the area of conceptual model development, direction of modeling,
26 and in writing the document. For this document, he co-authored Chapter 3.0 and reviewed the
27 approach, models, and results.

1 **Anthony J. Knepp**, Project Manager, YAHSGS, Inc.

> B.S. Engineering, Johns Hopkins University
> M.S. Environmental Systems Engineering, Clemson University

2 Mr. Knepp provided technical direction and guidance to the development of the performance
3 assessment approach, conclusion, and recommendations. Mr. Knepp was also responsible for
4 the production of two major supporting documents to this work: *Field Investigation Report*
5 *for Waste Management Area S-SX* and *Field Investigation Report for Waste Management*
6 *Area B-BX-BY*.

7 **Karrol D. Lehman**, Document Manager/Senior Technical Writer, Columbia Engineering and
8 Environmental Services, Inc.

> B.A. Biology, Central Washington University
> J.D. University of Puget Sound School of Law

9 Ms. Lehman has over 30 years of experience in regulatory programs, environmental compliance,
10 safety analysis and authorization basis documentation, and laboratory analyses. She has
11 provided technical and administrative expertise on regulatory compliance issues, authorization
12 basis management, regulatory inspections and audits, commitment tracking, and other topics
13 related to environmental compliance and radiological, nuclear, and process safety. She has
14 coordinated the production and assisted in the development and drafting of formal safety
15 documentation for Department of Energy facilities and has provided legal support, litigation
16 preparation, and case management support for a corporate legal department. She has conducted
17 training courses in various settings and is a Certified Hazardous Materials Manager.
18 Ms. Lehman served as document manager for this performance assessment, coordinating and
19 maintaining the document production schedule, as well as providing oversight for technical
20 writing and editing support.

21 **Frederick M. Mann**, Scientist, CH2M HILL Hanford Group, Inc.

> B.S. Physics, Stanford University
> Ph.D. Physics, California Institute of Technology

22 Dr. Mann has worked for over 25 years in the field of nuclear data and the application of those
23 data to large energy facilities, and has advised the Department of Energy and the International
24 Atomic Energy Agency. He produced the *Preliminary Performance Assessment for Waste*
25 *Management Area C at the Hanford Site, Washington*. Dr. Mann was also the team leader for
26 the immobilized low-activity waste performance assessment activity, which produced the 1998
27 and 2001 versions of the *Hanford Immobilized Low-Activity Tank Waste Performance*
28 *Assessment* and *Hanford Immobilized Low-Activity Tank Waste Performance Assessment*, as well
29 as the *Integrated Disposal Facility Risk Assessment* and the *Hanford Integrated Disposal*
30 *Facility Performance Assessment*. He led the effort that produced the *Risk Assessment*
31 *Supporting the Decision on the Initial Selection of Supplemental ILAW Technologies*.
32 Dr. Mann reviewed this document.

1 **William J. McMahon**, Scientist, CH2M HILL Hanford Group, Inc.

    B.S.   Agricultural Engineering, University of California, Davis
    M.S.   Agricultural Engineering, Texas A&M University

2 Mr. McMahon specializes in hydrologic data collection, analysis, and interpretation, and
3 groundwater and vadose zone numerical modeling to support groundwater and vadose remedial
4 projects. He has experience with RESRAD, MODFLOW, VAM3DCG, PORFLOW, STOMP,
5 and MICROFEM vadose and groundwater modeling packages. He also provides technical
6 direction, guidance, and oversight to subcontractors performing numerical modeling in support
7 of the risk assessments associated with accelerated retrieval and closure of single-shell tanks.
8 His other duties include directing hydrologic data collection efforts, analyzing and interpreting
9 hydrologic data, assessing the effectiveness of groundwater remedial actions, developing work
10 plans for data collection and interpretation, and performing numerical modeling to predict
11 facility impacts to the aquifer to support remediation and construction decisions. He co-authored
12 the sensitivity and groundwater contaminant analysis in Chapter 4.0.

13 **David A. Myers**, Geologist, CH2M HILL Hanford Group, Inc.

    B.S.   Geology, University of Puget Sound
    M.S.   Geology and Hydrology, University of Idaho

14 Mr. Myers is a registered professional geologist in Idaho and Oregon. His work has focused on
15 water resources, as well as environmental monitoring and remediation of groundwater
16 contamination. Since arriving at the Hanford Site in 1974, Mr. Myers has provided technical
17 support for the Site-Wide Groundwater Monitoring Program, as well as early development of the
18 RCRA monitoring program for the low-level waste burial grounds. He served as a senior
19 hydrogeologist within the environmental restoration program, actively participating in the design
20 and implementation of groundwater remediation projects. He supports the Tank Farm Vadose
21 Zone Project as a technical coordinator, ensuring that multiple aspects of this complex problem
22 are integrated and coordinated. He was responsible for preparing the facilities section of
23 Chapter 2.0 in this document.

24 **Angela M. Newell**, Research Specialist, Babcock Services, Inc.

    B.A.   English, Saint Edward's University
    B.A.   Theatre Arts, Saint Edward's University
    M.S.   Public Policy and Management, Carnegie Mellon University

25 Ms. Newell is a trained statistician and technology specialist. Over the past year at the Hanford
26 Site, she researched and developed a data management system for single-shell tank leak history
27 and characterization. For this document, Ms. Newell co-authored sections on human health and
28 the environment, compiled the SST performance assessment recovery and maintenance plans,
29 and provided analytical and general research assistance and document review.

1 **Raymond J. Puigh II**, Manager, Nuclear and Environmental Initiatives, Fluor Federal Services

> B.S. Physics, Louisiana State University
> Ph.D. Physics, Florida State University

2 Dr. Puigh has over 25 years of experience in nuclear fields ranging from nuclear data and
3 materials testing to nuclear safety and facility authorization bases analyses. Dr. Puigh has over
4 10 years experience with performance assessments including metallurgy, tank waste inventory
5 support, safety assessment support for selected tank farm operation projects, and management of
6 environmental modeling staff responsible for the Hanford Site solid waste, immobilized
7 low-activity waste, integrated disposal facility, and tank farm performance assessments. He is
8 one of the main authors for the 2001 Immobilized Low-Activity Waste and the 2005 Integrated
9 Disposal Facility performance assessments. He co-authored Chapter 3.0, and provided reviews
10 and comments on other chapters.

11 **Paul D. Rittmann**, Process/Specialty Engineer II, Safety Analysis and Nuclear Engineering,
12 Fluor Government Group, Inc.

> B.A. Physics, University of Wisconsin
> M.S. Physics, Purdue University
> Ph.D. Physics, Purdue University

13 Dr. Rittmann has been employed at the Hanford Site since 1980 and has been using
14 environmental transport and dose models since 1983. He was a member of the Hanford
15 Environmental Dose Overview Panel and the American Academy of Health Physics. He was
16 responsible for the dosimetry analysis for the Hanford Site solid waste and immobilized
17 low-activity waste performance assessments. Dr. Rittman supplied the dosimetry data and
18 analysis for this performance assessment.

19 **Craig A. Sump**, Scientist, CH2M HILL, Inc.

> B.S. Geology, Washington State University
> B.A. Philosophy, Washington State University

20 Mr. Sump has over 8 years experience as a geologist and environmental consultant involving
21 mining, mine reclamation and acid rock drainage remediation, regulatory compliance, surface
22 water management, contaminated groundwater investigations, and site characterization activities.
23 His focus over the past 4 years has been to support site characterization and remediation
24 activities at the Department of Energy Hanford Site. For this performance assessment, he
25 contributed to the development of the Decision Management Tool software, and assisted in
26 compiling and interpreting groundwater concentration fate and transport modeling results
27 presented in Chapter 4.0.

1 **David J. Watson**, Scientist, CH2M HILL Hanford Group, Inc.

      B.S.   Geology, Washington State University

2 Mr. Watson has over 9 years experience in data management and interpretation, the last 5 of
3 which have been related to site characterization and remediation activities at the Department of
4 Energy Hanford Site. For the last 4 years, he has developed the Decision Management Tool,
5 a risk assessment software package for the CH2M HILL Hanford Group, Inc. Tank Farm Vadose
6 Zone Project. He is also responsible for supporting tank closure risk assessments, calculating
7 contaminant groundwater concentrations and related risks to human health, as well as providing
8 data analysis and interpretation. He prepared the quality assurance sections in Appendix F,
9 coordinated preparation of the reference case groundwater modeling results sections of
10 Chapter 4.0, and co-authored Chapters 4.0 and 6.0.

11 **Marcus I. Wood**, Principal Scientist, Waste Management, Fluor Hanford, Inc.

      B.S.   Geology, University of North Carolina
      Ph.D. Geology, Brown University

12 Dr. Wood currently is responsible for developing and maintaining the performance assessment
13 analyses for the disposal of solid low-level waste at the Hanford Site. He is the coordinating
14 author of the Hanford Site solid waste performance assessments and has been largely responsible
15 for the integration and the interpretation of the analytical results in those documents. He also
16 contributes to the development of field characterization studies of soils contaminated by
17 single-shell tank waste in SST waste management areas, and has developed conceptual models of
18 subsurface migration of tank waste by integrating historical and recent characterization data.
19 He is a primary author of documents analyzing the impacts of further tank waste migration on
20 the surrounding environment. He has also coordinated performance assessment analyses for the
21 Environmental Restoration Disposal Facility (ERDF), at which wastes generated in the
22 remediation of Hanford Site waste sites regulated under the CERCLA are disposed. He has
23 directed numerous projects to quantify the geochemical properties of radionuclides in the
24 Hanford Site geohydrologic environment, and was responsible for developing a multifunctional
25 waste package backfill material for isolating spent fuel and high-level waste. He prepared
26 discussions concerning the defense in depth philosophy and sensitivity and uncertainty analyses.
27 Dr. Wood supported the development of the approach and co-authored the sensitivity analysis in
28 Chapter 4.0.

1 **Fred Zhang**, Senior Research Scientist, Hydrology Group, Pacific Northwest National
2 Laboratory

        B.S.   Soil and Agro-Chemistry, Beijing Agricultural University
        M.S.   Soil Science, Beijing Agricultural University
        Ph.D.   Soil Physics, University of Guelph
                Diploma of Micro-Computer Systems, McMaster University

3 Dr. Zhang has expertise in the calibration of flow and transport model using inverse techniques
4 and has a strong background in mathematics and computer programming. He is currently
5 involved in the Vadose Zone Transport Field Study at the Hanford Site as part of the Science and
6 Technology Program. For this project, he performed the numerical simulations of contaminant
7 transport for the vadose zone modeling and groundwater pathway effort.