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LITER DOSES FOR TWRS FSAR RADIOLOGICAL
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Development of Radiological Concentrations and Unit Liter Doses for TWRS FSAR Radiological Consequence Calculations

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Abstract: The analysis described in this report develops the Unit Liter Doses for use in the TWRS FSAR. The Unit Liter Doses provide a practical way to calculate conservative radiological consequences for a variety of potential accidents for the tank farms.

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EXECUTIVE SUMMARY

Detailed information pertaining to a key element in WHC-SD-WM-SAR-067, *Tank Waste Remediation System (TWRS) Final Safety Analysis Report (FSAR)** accident analysis - unit liter dose (ULD) - is provided in this report. (ULDs are the radiation doses received per liter of tank waste, from direct radiation, inhalation, and 24 hour ingestion and are used to support the consequence analysis of the TWRS FSAR.) The concentrations and ULDs make best use of available data. ULDs along with the volume release fraction and atmospheric dispersion factor are the key elements of the accident analysis source term. This report shows the technical basis justifying that selected values are conservative.

The basis for a comprehensive (physical scope and important radionuclides) set of bounding input information for the TWRS FSAR is presented in this report. The ULDs are used in the FSAR to calculate potential accident consequences for release of wastes from the 177 high-level nuclear waste tanks and transfer piping. Some descriptions of the TWRS inactive miscellaneous underground storage tanks (IMUSTs) are also included. The interfaces with criticality specifications, the 204-AR Waste Unloading Facility, and the 242-A Evaporator are also addressed. The rationale for selection of the tank waste models, specific radionuclides, activity concentrations, and derived ULDs are explained.

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*WHC-SD-WM-SAR-067, 1996, *Tank Waste Remediation System (TWRS) Final Safety Analysis Report*, DRAFT, Westinghouse Hanford Company, Richland, Washington.

The analysis develops eight waste models to balance conservatism, safety analysis complexity, and uncertainty. Model selection considers the nonhomogeneous nature of the waste and constraints on data. The eight models separate the physical waste phases (liquid versus solid), waste characteristics (aging waste versus other waste), and tank physical characteristics (single-shell tanks [SSTs] and double-shell tanks [DSTs]). Two additional waste models address specific analytical needs to 'bin' tanks. Each waste model combines data from all of the tanks represented by the model.

Subject matter experts derived radionuclide activity concentrations from sample data. In the cases where sample data was not available, the experts used knowledge of reactor output and known relationships with measured radioisotopes. The experts created a database of more than 11,000 data points to support sample-based selection of activity concentrations. The database includes both analytical sample results and data calculated by process models. The experts screened the database using a structured process to evaluate and select activity concentrations.

The experts evaluated more than 150 radionuclides stored in the Hanford Tanks. ULD calculations use 22 of the 150 plus radionuclides. The 22 radionuclides account for nearly all of the dose hazard. The 11 radionuclides that comprise 99+% of the inhalation dose are presented in this report for simplicity. Conservatism in the method and data uncertainty justify this simplification because both are far larger than 1%.

Consequence analyses consider direct radiation, ingestion and inhalation doses. Emergency preparedness actions can interdict food and water consumption within 24 hours, so the ULDs do not assess ingestion beyond 24 hours. Environmental impact statements address the consequences of long-term releases to the ground, groundwater, and the environment. Source terms to support the Environmental Impact Statement process are not addressed in this report.

The ULDs for each of the waste models are presented in the table below. For each isotope, calculation used the highest verified data point from the tanks represented by the model. Statistical evaluation demonstrates this method successfully selects the highest valid data point 95% of the time with 95% confidence for each isotope. The method for calculating dose consequences using the ULDs presented in this document is given in WHC-SD-WM-SARR-016, *Tank Farm HLW Compositions and Atmospheric Dispersion Coefficients for Use in ASA Consequence Assessments*.^{*} The ULDs provide an efficient method to account for multiple radionuclides and waste types.

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^{*}WHC-SD-WM-SARR-016, 1995, *Tank Farm HLW Compositions and Atmospheric Dispersion Coefficients for Use in ASA Consequence Assessments*, Westinghouse Hanford Company, Richland, Washington.

Unit Liter Doses for Both Inhalation and Ingestion.

Composite	Inhalation ULD (Sv/L)	Ingestion ULD* (Sv-m ³ /s-L)
All liquids	1.2 E+04	0.11
All solids	1.9 E+06	8.2
Single-shell tank liquids	1.1 E+04	0.052
Single-shell tank solids	2.2 E+05	4.1
Double-shell tank liquids	6.1 E+03	0.068
Double-shell tank solids	5.3 E+05	0.48
Aging waste facility liquids	1.4 E+03	0.092
Aging waste facility solids	1.7 E+06	8.1

*Includes 24-hours ingestion of fruits and vegetables, ground shine, inadvertent soil ingestion, and inhalation of material resuspended from the ground.

Sv/L = sieverts per liter.
 $\text{sv-m}^3/\text{s-L} = \frac{\text{sieverts} \cdot \text{cubic meters}}{\text{seconds} \cdot \text{liter}}$
 ULD = unit liter doses.

The analysis verified conservatism and reasonableness of the activity concentrations several ways.

1. The analysis compared total activity of each radionuclide used to calculate ULDs with total activities for the same radioisotopes calculated from total isotope production in the Hanford Site reactors. The comparison shows individual isotope ULDs ranging from 7 to over 1,473 times the production-based values, and a weighted average about 58 times the reactor-based value. (This was expected because of the data distribution and concentration selection method.)
2. The overall selection methodology is directly conservative because it selects the high points of a sample distribution rather than

average values, the maximum ULD for any tank represented by the model, or other less conservative statistical values.

3. A statistical analysis shows that the probability of finding a higher value for a radionuclide in the database is less than 4%, with 95% confidence. This result provides high confidence that individual tanks or waste volumes released in the unlikely event of an accident will not exceed the safety analysis values.
4. An analysis for SSTs shows that unit heat loads calculated using the activity concentrations selected for the ULDs far exceed, on the average, the unit heat loads calculated using actual in-tank temperatures.
5. Sample data and selected concentrations are consistent with known processes and expectations.

The report recommends ULDs for other facilities and interfaces, where available. It describes IMUSTs, but does not present ULDs since analysts could find no relevant data. The tables present ULDs for the transfer lines, 204-AR facility, interfaces with the evaporator, and the interface with the criticality specification.

The report highlights key assumptions that may lead to a need for safety controls. The conclusions of this report demonstrate that this analysis satisfies all objectives.

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LIST OF TERMS

AWF	aging waste facility
DST	double-shell tank
FSAR	final safety analysis report
ICF KH	ICF Kaiser Hanford Company
IMUST	Inactive Miscellaneous Underground Storage Tank
PNNL	Pacific Northwest National Laboratory
PUREX	Plutonium-Uranium Extraction (Facility)
REDOX	reduction and oxidation
SST	single-shell tank
TRAC	track radioactive components
TSR	Technical Safety Requirement
TWRS	Tank Waste Remediation System
ULD	unit liter dose
WHC	Westinghouse Hanford Company

1.0 INTRODUCTION

1.1 PURPOSE

The purpose of the analysis presented in this report is to support safe operation of high-level nuclear waste tanks at the Hanford Site by deriving conservative values of the radionuclide concentrations that are broadly applicable to waste chemistry and tank unit liter doses (ULDs). (ULDs are the radiation dose received from a liter of tank waste, including direct radiation, inhalation, and 24-hour ingestion.) Bounding (high) ULDs ensure conservative estimates of risks from the Hanford tanks. The analysis process makes best use of available data and substantiates the basis for data selection. Waste models, selected radionuclides, activity concentrations, and ULDs for WHC-SD-WM-SAR-067, *Tank Waste Remediation System (TWRS) Final Safety Analysis Report (FSAR)*, are provided in this report.

Westinghouse Hanford Company (WHC) established the following objectives for the process to develop concentrations and ULDs:

1. Specify the radionuclide concentrations and ULDs for the TWRS safety analysis.
2. Include all large underground storage tanks (single-shell tanks [SSTs] and double-shell tanks [DSTs]).
3. Substantiate conservative values for concentrations and ULDs.
4. Develop a logical method to "bin" (or group) the tanks.
5. Identify assumptions TWRS must protect to ensure safety analysis validity.

This document is part of a technical baseline that provides the comprehensive basis for TWRS safety. Placement of this document in a hierarchy of directly related documents is shown in Figure 1. This document supplements and updates information presented in WHC-SD-WM-SARR-016, *Tank Farm HLW Compositions and Atmospheric Dispersion Coefficients for Use in ASA Consequence Assessments*. The TWRS FSAR is the top document in this hierarchy. The TWRS FSAR is the basis for setting the objectives for this analysis and is the primary 'customer' for the analysis results.

1.2 UNIT LITER DOSE DEVELOPMENT PROCESS

The analysis described in this report develops the ULDs for use in the TWRS FSAR. The ULDs provide a practical way to calculate conservative radiological dose consequences for a variety of potential accidents.

Safety analysis relies on consequence analysis to calculate the radiation dose to defined receptors onsite and offsite. These analyses need a 'source term' to calculate the dose. The source term is a quantity of specified hazardous material. The material specification must include quantity,

Figure 1. Document Hierarchy of Comprehensive Technical Basis Documentation for TWRS FSAR.

^a WHC-SD-WM-SARR-016, 1995, *Tank Waste Compositions and Atmospheric Dispersion Coefficients for Use in ASA Consequence Assessments*, Rev.1, Westinghouse Hanford Company, Richland, Washington.

^b PNL 6584, 1988, *GENII-Hanford Environmental Radiation Dosimetry Software System*, Pacific Northwest Laboratory, Richland, Washington.

^c WHC-SD-WM-ER-400, 1995, *Tank Waste Source Term Inventory Validation*, Rev 0, Volumes I and II, Westinghouse Hanford Company, Richland, Washington.

TWRS = Tank Waste Remediation System.
FSAR = Final Safety Analysis Report.

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physical form, and specific quantities of hazardous material. For radiation hazard, this includes specifying the particle sizes, quantity in kilograms per volume of air, radioactive isotopes present, and concentrations of the isotopes.

The overall logic used to develop the ULD concentrations and ULDs, and to assess the conservatism of the result is presented in Figure 2. The process of developing and assessing ULDs is described in Section 3.0, with section numbers that correspond to the process block numbers on Figure 2 (i.e., Section 3.1 corresponds to process block 1 of Figure 2).

Where possible, the analysis uses sample data to develop the radionuclide activity concentrations. In the cases where sample data is not available, analysts developed concentrations using knowledge of reactor output and known relationships of the subject radionuclide with other measured radioisotopes. To support the sample-based selection of activity concentrations, WHC and ICF Kaiser Hanford Company (ICF KH) created a database that included over 11,000 data points. The database includes both analytical sample results and data calculated by process models.

Several tests demonstrated that the selected concentrations are at or near the upper limit of any samples taken from the tanks. The tests included the following:

- Comparison of total curies based on a maximum sample versus ORIGEN2 (ORNL-5621) data
- Evaluation of process history
- Comparison of heat loads calculated from nonsample data
- Statistical evaluation of the data.

The analysis initially developed ULDs to support consequence analysis based on inhalation dose, and for one class of accidents that form surface pools which might deliver direct radiation to a facility worker. Emergency action can interdict food and water in the short term. The consequence analysis method uses the ULDs to include ingestion and ground shine during the first 24 hours to allow for time to establish interdiction. DOE/EIS-0113, *Final Environmental Impact Statement: Disposal of Hanford Defense High-Level, Transuranic and Tank Wastes, Hanford Site, Richland, Washington*, assesses long-term consequences (50 years) and environmental impact of ground and groundwater releases.

This report presents 11 radionuclides for each ULD to simplify the presentation. The actual ULD calculations use 22 radionuclides to account for nearly the entire dose. The 11 radionuclides represent over 99% of the radioactive dose calculated using all isotopes (over 150) evaluated by the ORIGEN2 computer code.

1.3 SCOPE

The facilities and interfacing facilities included in the scope of this document are listed in Table 1.

Figure 2. Logical Process of ULD Development.

WHC-SD-WM-ER-400, 1995, *Tank Waste Source Term Inventory Validation*, Rev 0.
 Volumes I and II, Westinghouse Hanford Company, Richland, Washington.

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Table 1. TWRS FSAR Radiological Sources and Interfacing Facilities.

Facilities included in scope	Interfacing facilities
Single-shell tanks	None
Double-shell tanks	Single-shell tanks, 242-A Evaporator, AWF tanks
Aging waste facility tanks	Single-shell tanks, 242-A Evaporator, double-shell tanks
200 series single-shell tanks	None
Inactive miscellaneous underground storage tanks	None
204-AR	Double-shell tanks, waste generators

FSAR = final safety analysis report.
 TWRS = Tank Waste Remediation System.

This document does not include tank waste retrieval projects. The ULDs in this document have not been evaluated for retrieval.

1.4 DOCUMENT USER GUIDANCE

The facilities to which the concentrations and ULDs in the TWRS FSAR apply are discussed in Section 1.3. Assumptions listed in Section 5.0 are for use in the TWRS safety analysis process to identify Technical Safety Requirements (TSRs), or other safety controls.

The ULDs in this report apply to accidents that expel some quantity of waste from the tanks in the physical and chemical form in which it exists in the waste tanks. For accident conditions that change the chemical or physical form of the waste, the accident analysis must determine the effective release quantities of the component radionuclides.

TWRS will establish a process to evaluate new sample data. The process must evaluate radionuclide concentrations above the concentrations provided herein to determine the potential impact on the TWRS authorization basis.

1.5 DOCUMENT ORGANIZATION

General information about the wastes and facilities addressed by this report and how this report supports the overall consequence analysis process is provided in Section 2.0.

The entire ULD development and verification process is described in Section 3.0. This process includes the following: selecting the activity concentrations; determining radionuclide concentrations when no sample results were available; and verifying that selected activity concentrations represented bounding values.

Subsection numbers correspond to the process blocks illustrated in Figure 2. The ULDs, selected activity concentrations, and the percentage contribution to total dose of each of the 11 major radionuclides are presented in tables.

The interface between the tanks and other facilities is described in Section 4.0. Considerations for TSR development are presented in Section 5.0. The conclusions are given in Section 6.0.

2.0 BACKGROUND INFORMATION

2.1 HANFORD TANKS

Multiple waste processes and limited access to the waste cause two problems. Multiple waste processes create nonhomogeneous waste and uncertainty about waste composition. Limited access reduces the representativeness of tank characterization data.

Hanford Site wastes in the high-level nuclear waste tanks are not uniform in either composition or distribution. Wastes are similar in that they are moderately to highly basic and have high concentrations of hydroxides, nitrate salts, and water. Separations plants used significantly different processes to generate the wastes. After discharge from the separations plants to the tanks, waste was precipitated and evaporated by different processes to reduce volume, allowing more waste storage. Waste was pumped from tank-to-tank and farm-to-farm. Therefore while the same radionuclides may be in all the tanks, radionuclide concentrations vary from tank-to-tank.

The characterization program, described in WHC-SD-WM-TA-163, *Tank Waste Characterization Process*, can only access the waste in the tanks through vertical metal pipes called risers. The risers range through 5 cm (2 in.) in diameter to 107 cm (42 in.) in diameter. The risers limit the locations of samples that can be taken from a tank. This makes uniform sampling of the tank contents very difficult. Using a large sample database and interpreting process knowledge and operations history helps to ensure identification of representative data.

2.2 SPECIFIC FACILITIES

2.2.1 100-Series Tanks

There are 161 tanks with volumes ranging from 1,893 kL to 3,785 kL (500,000 to more than 1,000,000 gal), which are sometimes called the 100-series tanks.

This report breaks the 100-series tanks into three categories: SSTs, DSTs, and aging waste facility (AWF) tanks. The distinction between SSTs and DSTs is based on the construction of the tanks. The SSTs are a concrete

cylinder with an internal steel liner that covers only the bottom and sides of the cylinder. The DSTs are a concrete cylinder with a steel liner that forms a complete steel cylinder within the concrete cylinder. DSTs also have a second steel liner that forms a cup. The AWF tanks are a subset of the DSTs. Four tanks are designated AWF tanks because they are capable of storing first-cycle waste, which contain most of the mixed fission products discharged from the Plutonium Uranium Extraction (PUREX) Facility during the 1980's. The tanks are grouped into farms, ranging from two tanks per farm to 18 tanks per farm.

2.2.2 200-Series Tanks

There are 16 SSTs with volumes of 55,000 gallons each. Four such SSTs are located in each of B, C, T, and U Tank Farms. These tanks also contain process waste. Structurally the tanks are similar to the larger 100-series SSTs. These 16 200-series SSTs plus the 161 100-series tanks equal the commonly referred to total of 177 storage tanks at the Hanford Site.

2.2.3 Inactive Miscellaneous Underground Storage Tanks

The inactive miscellaneous underground storage tanks (IMUSTs) were used for many functions associated with plutonium and uranium separations and waste management activities in the 200 East and 200 West Areas of the Hanford Site. The IMUSTs include settling tanks, process tanks, catch tanks, and chemical storage. At different times, some IMUSTs contained radioactive process wastes. Only the 36 IMUSTs assigned to TWRS (WHC-EP-0861, *Status Report for Inactive Miscellaneous Underground Storage Tanks at the Hanford Site 200 Areas*) are included in this report; however, additional facilities may be identified as IMUSTs in the future.

Most of the IMUSTs contain some wastes and are interim isolated and interim stabilized. Interim isolated means the tanks are isolated from facilities by sealing all inlet and outlet connections to the tanks. Interim stabilized means that all liquid that is technically and economically practical to remove has been removed. The total remaining volume of liquids and solids in the IMUSTs is approximately 0.4% of the volume of the wastes in the SSTs (WHC-EP-0861).

2.3 CONSEQUENCES ANALYSIS PROCESS

This report is part of the overall consequence analysis process, which is outlined in Figure 3. The process begins with selection of an accident scenario and ends with the calculations of the radiological dose to an individual. The following three parts of the analysis process are presented in this report.

- Block 2, "Select waste groupings"
- Block 3, "Determine source isotopes and concentrations"
- Block 4, "Calculate ULDs."

Figure 3. Generalized Safety Analysis Model of Basis for Defining Concentration Accuracy Requirements.

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For important considerations in calculating total dose shown in Figure 3 but not covered in this document (e.g., calculating the quantity of material released, the transport model for the material, and the location of the exposed individual), the reader should refer to the TWRS FSAR.

2.3.1 Ingestion and Inhalation Doses

The ingestion dose and inhalation dose can be combined in calculating consequences to the offsite receptor. The ingestion dose is:

$$\text{Dose (ingestion)} = Q * \text{ULD (ingestion)} * X/Q.$$

The dose from inhalation is given by:

$$\text{Dose (inhalation)} = Q * \text{ULD (inhalation)} * X/Q * \text{BR}$$

where:

- Q = amount released (liters)
- ULD (inhalation) = unit liter dose in sieverts per liter (Sv/L)
- ULD (ingestion) = unit liter dose in Sv-m³/L-s
- X/Q = atmospheric dispersion coefficient in seconds per cubic meter (s/m³)
- BR = breathing rate in cubic meters per second (m³/s).

These two ULDs are used differently because the inhalation requires multiplying by breathing rate. The total dose is therefore:

$$\begin{aligned} \text{Dose(total)} &= \text{Dose(inhalation)} + \text{Dose(ingestion)} \\ &= Q * X/Q * [\{\text{ULD(inhalation)} * \text{BR}\} + \text{ULD(ingestion)}]. \end{aligned}$$

The inhalation ULDs in Table 2 are developed in WHC-SD-WM-SARR-016. Usually, including ingestion increases the offsite doses slightly. For an acute breathing rate of 3.3×10^{-4} m³/s, including the ingestion dose increases the total dose from 0.3 to 5% in all composites except the AWF liquids, which show a 20% increase.

2.3.2 Plume Submersion Doses

ULDs for the eight waste models do not explicitly include plume submersion doses (the phenomenon of being surrounded by a semi-infinite cloud of released material). Knowledge of plume submersion calculations demonstrates that plume submersion doses are significant only for noble gases (e.g., krypton and xenon). These isotopes are not present in the tank wastes because of their relatively short half lives and partitioning during chemical processing.

The inhalation dose conversion factors are high for ⁹⁰Sr and transuranics, further ensuring that any direct dose from the plume would be negligible.

Table 2. Unit Liter Doses for both Inhalation and Ingestion.

Composite	Inhalation ULD (Sv/L)	Ingestion ULD* (Sv-m ³ /s-L)
All liquids	1.2 E+04	0.11
All solids	1.9 E+06	8.2
Single-shell tank liquids	1.1 E+04	0.052
Single-shell tank solids	2.2 E+05	4.1
Double-shell tank liquids	6.1 E+03	0.068
Double-shell tank solids	5.3 E+05	0.48
Aging waste facility liquids	1.4 E+03	0.092
Aging waste facility solids	1.7 E+06	8.1

*Includes 24-hours ingestion of fruits and vegetables, ground shine, inadvertent soil ingestion, and inhalation of material resuspended from the ground.

Sv/L = sieverts per liter.

sv-m³/s-L = $\frac{\text{sieverts} \cdot \text{cubic meters}}{\text{seconds} \cdot \text{liter}}$.

ULD = unit liter doses.

3.0 HIGH-LEVEL WASTE TANKS

WHC safety analyses use ULDs to calculate accident consequences. ULD values are used to characterize the tank waste in terms of its inhalation, direct shine, and ingestion hazard. Inhalation dose is given in units of committed effective dose equivalent (CEDE) per liter of waste inhaled. Ingestion doses are given for a ULD release based on an atmospheric dispersion coefficient (X/Q) of 1.0.

Analysts calculated each ULD using a different waste model. The waste model approach is a conservative way to evaluate accident dose consequences using the available tank waste characterization data. The model maximizes dose consequences by figuratively placing into a single hypothetical tank the highest activity concentration of every radionuclide (i.e., ⁹⁰Sr, ¹³⁷Cs, and ²³⁹Pu) contained in the samples for all of the tanks in that waste model. Using a model tank minimizes the potential for nonconservatism associated with the use of tank-by-tank concentrations when tank inventories change. It also minimizes the confusion associated with the use of numerous tank-by-tank concentrations while providing some credit for the variability of the tank waste hazard through tank subgroups. The model tank ULDs have been generated in such a way as to always bound the maximum tank-by-tank concentrations.

3.1 SELECT NUCLIDES

The process used to select the nuclides important to dose is illustrated in more detail in Figure 4. (This is an expansion of process block 1 on Figure 2.) The process selectively reduces the total number of radionuclides

Figure 4. Key Nuclides Starting With ORIGEN2 Isotope List of >150 Isotopes (Eleven Nuclides Account for >99% of Dose).

^a ORNL-5621, *ORIGEN2-A Revised and Updated Version of the Oak Ridge Isotope Generation and Depletion Code*, 1980, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

^b EPA-520/H/88-020, 1988, *Limiting Values of Radionuclide-Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion, Federal Guidance Report No. 11*, U.S. Environmental Protection Agency, Office of Radiation Programs, Washington, D.C..

^c For historical reasons, an additional 11 nuclides are used in the unit liter dose calculations. These add less than 1% to total dose.

DCF = dose conversion factor

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to the 11 addressed in this report. (Although an additional 11 radionuclides are included in the calculation of ULDs for historical reasons, their contribution to dose is insignificant.) The process started with evaluation of more than 150 nuclides tracked by the ORIGEN2 computer code (ORNL-5621).

3.1.1 Radionuclide Identification

FSARs report accident consequences in terms of the dose to the onsite worker and the public. Although many radionuclides can be detected in Hanford Site waste, the number of radionuclides that are significant for radiological dose calculations is limited. For the analyses in the FSAR, a group of 11 radionuclides was identified that accounts for 99+% of the radiological dose. The 11 radionuclides include the following: ^{60}Co , ^{90}Sr , ^{90}Y , ^{137}Cs , ^{154}Eu , ^{23}Np , ^{238}Pu , $^{239/240}\text{Pu}$, ^{241}Pu , ^{241}Am , and ^{244}Cm .

The selected radionuclides are not equally significant for each of the eight models identified in the next section of this report. For some of the ULDs, four or five radionuclides (^{90}Sr , ^{90}Y , ^{137}Cs , ^{239}Pu , and ^{241}Am) make up about 95% of the total dose. However, for consistency, the same 11 radionuclides were used for each model tank.

Several waste transfer accident scenarios result in spills of radioactive material to the ground. These spills are a source of concentrated radioactivity and result in a direct external dose to workers. The dominant contributors to the direct external dose (^{90}Sr , ^{90}Y , and ^{137}Cs) are also included in the 11 radionuclides selected because of their high contribution to inhalation dose. Two of the 11 radionuclides were selected (^{60}Co and ^{154}Eu) because of their contribution to the direct external dose although their contribution to the inhalation dose is less than 1%. The process used to identify the key dose contributors is described below.

3.1.1.1 ORIGEN2 Data. The ORIGEN2 computer code (ORNL-5621) was used to estimate the inventory of radioactive materials contained in the fuel from the single-pass reactors and N Reactor, as reported in WHC-SD-WM-RPT-164, *Inventories for Low-Level Waste Tank Waste* (see Figure 4, block 1-1). Total radionuclide inventories were calculated based on the complete operating history of all of the Hanford Site production reactors. Reduction factors were then applied to the total inventories to account for plutonium and uranium extracted from the waste sent to the tanks. Reduction factors also were applied to other isotopes, including cesium and strontium, which also were extracted from the waste.

The use of ORIGEN2 computer code (ORNL-5621) in radionuclide development is shown in Figure 5. The model predicted the total radioactive inventory that existed in the SSTs and DSTs at the end of 1990. A separate ORIGEN2 run was made to decay the radioactive inventory in WHC-SD-WM-RPT-164 to the end of 1994 so that a comparison could be made to the values in WHC-SD-WM-ER-400, *Tank Waste Source Term Inventory Validation*, which are also decayed to the end of 1994.

Figure 5. Total Hanford Site Production History Used to Develop the Important Radionuclides.

ORNL-5621, 1980, *ORIGEN2-A Revised and Updated Version of the Oak Ridge Isotope Generation and Depletion Code*, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

PNL 6584, 1988, *GENII-Hanford Environmental Radiation Dosimetry Software System*, Pacific Northwest Laboratory, Richland, Washington.

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The ORIGEN2 computer code (ORNL-5621) estimates activity inventories for three different groups: actinides, fission products, and activation products. The total activity for each radionuclide (Table 3) from ORIGEN2 was multiplied by the inhalation dose conversion factor to determine the inhalation dose hazard for that radionuclide. The dose conversion factors were taken from EPA-520/1-88-020, *Federal Guidance Report No. 11*, and represent the committed effective dose equivalent in units of sieverts per becquerel for a 50-year dose commitment period.

ORIGEN2 computer code (ORNL-5621) calculations identified seven radioisotopes that contribute approximately 99% of the inhalation dose. Analysts added two radionuclides (^{237}Np and ^{244}Cm) based on process knowledge and two additional radionuclides (^{60}Co and ^{154}Eu) to account for direct radiation. Four of the original seven radionuclides (^{241}Am , ^{90}Sr , ^{137}Cs , ^{239}Pu) contribute 95% of the inhalation dose (Figure 6). The detailed derivation of the activity concentrations for the four radionuclides is given in Section 3.3.2. Additional information is provided in Appendix C.

3.1.1.2 Comparison of Relative Inhalation Hazard Based on ORIGEN2 Computer Code. The radionuclides were then ranked by inhalation dose hazard as shown in Table 3 and Figure 4, block 1-2. The 11 radionuclides listed in Table 3 make up 99% of the total dose calculated by the ORIGEN2 computer code (ORNL-5621). ULDs use an additional 11 radionuclides (see WHC-SD-WM-SARR-016), for a total of 22 radionuclides. Because the second group of 11 radionuclides collectively added <1% to the total ULD, they are not addressed in this report.

Subject matter experts established maximum sample activity concentrations for each of the 11 radionuclides included in the FSAR. Radionuclides were ranked by a process, similar to that done using activity inventories from ORIGEN2 computer code calculations (ORNL-5621), using the FSAR source term activity concentrations. The "FSAR total activity" column in Table 3 was developed by multiplying activity concentrations reported in the FSAR for the six primary tank waste types (i.e., SST liquids and solids, DST liquids and solids, AWF liquids and solids) by the tank waste volume associated with that waste type. The tank waste volumes were taken from WHC-EP-0182-81, *Waste Tank Summary Report for Month Ending December 31, 1994*.¹ This resulted in total activity inventories for each of the six waste types, which were then added to obtain the FSAR total activity, as reported in Table 3.

3.1.1.3 Measurement Accuracy. The accuracy and precision of measurements were considered in developing radionuclide concentrations and ULDs. The database accepted existing sample data from a variety of sources, subject to the following criteria: (1) sampling was done to meet a variety of process control and regulatory needs; (2) sample workup, assay procedures, and equipment used for analysis varied; and (3) access to waste was limited by riser configurations, which also often represented the points where waste was added to the tanks, making these points less representative of the tanks as a whole.

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¹It is recognized that sometimes there are differences in reported volumes between WHC-EP-0182-81 and the Waste Tanks Safety Program due to a time lag between measurement and publication.

Table 3. Comparison of ORIGEN2* and FSAR Total Inventories and Inhalation Dose Contributions.

Nuclide	ORIGEN2 total activity (Ci)	Dose conversion factor (rem/Ci)	ORIGEN2 dose (rem)	ORIGEN2 percent total dose	FSAR total activity (Ci)	FSAR dose (rem)	FSAR percent total dose	FSAR dose/ ORIGEN2 dose
⁶⁰ Co	9.24 E+03	2.19 E+05	2.02 E+09	0.0	1.55 E+06	3.38 E+11	0.0	167
⁹⁰ Sr	8.64 E+07	2.39 E+05	2.06 E+13	30.3	5.85 E+09	1.40 E+15	35.5	68
⁹⁰ Y	8.64 E+07	8.44 E+03	7.29 E+11	1.1	5.85 E+09	4.93 E+13	1.3	68
¹³⁷ Cs	7.16 E+07	3.19 E+04	2.28 E+12	3.4	4.94 E+08	1.57 E+13	0.4	7
¹⁵⁴ Eu	2.32 E+05	2.86 E+05	6.64 E+10	0.1	2.14 E+07	6.13 E+12	0.2	92
²³⁷ Np	7.45 E+01	5.40 E+08	4.02 E+10	0.1	1.10 E+05	5.93 E+13	1.5	1,473
²³⁸ Pu	2.52 E+03	3.92 E+08	9.88 E+11	1.5	7.28 E+05	2.85 E+14	7.3	289
²³⁹ Pu	4.43 E+04	4.29 E+08	1.90 E+13	27.9	2.39 E+06	1.03 E+15	26.1	54
²⁴¹ Pu	1.48 E+05	8.25 E+06	1.22 E+12	1.8	1.35 E+07	1.12 E+14	2.8	91
²⁴¹ Am	5.18 E+04	4.44 E+08	2.30 E+13	33.8	2.20 E+06	9.78 E+14	24.9	43
²⁴⁴ Cm	4.04 E+02	2.48 E+08	1.00 E+11	0.1	1.63 E+04	4.05 E+12	0.1	40
Total			6.48 E+13			3.93 E+15		

*ORNL-5621, 1980, ORIGEN2 - A Revised and Updated Version of the Oak Ridge Isotope Generation and Depletion Code, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

FSAR = final safety analysis report.

Figure 6. Nine Isotopes Selected for Inhalation Dose Deliver 99+% of Total Inhaled Dose (Four of the Nine Deliver 95% of the Total Inhalation Dose).

The nine isotopes selected for inhalation dose deliver 99+% of total inhaled dose. Four of the nine deliver 95%.

Note: Total is greater than 100% because of rounding

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The reported highest value for each waste sample was selected; i.e., assays did not add uncertainty to the value. The reason is that the best known assay uncertainty (counting in the laboratory) is insignificant compared to the actual variation in the waste.

For example, WHC-SD-WM-DTR-026, *Laboratory Characterization of Samples Taken in December 1991 (Window E) from Hanford Waste Tank 241-SY-101*, reports sample uncertainties as follows:

Random Error [(Variance/Mean) x 100] for Typical Samples		
	Liquid	Solid
¹³⁷ Cs	8.5%	4.9%
²⁴¹ Am	13.3%	10.0%

In contrast, WHC-SD-WM-ER-400 scatter plots show radionuclide data ranges from minimum to maximum of two to more than three orders of magnitude. Therefore, the data variability caused by sampling and inhomogeneity is on the order of 300 to 3,000 times the laboratory uncertainty (see Appendix D for sample distributions). The variance due to inhomogeneity in the waste is included in the variance observed in the sample set.

The subject matter experts reviewed all data. Experts included experienced personnel from the analytical laboratory, process chemistry, TWRS engineering, waste tank operations, and safety analysis. They used the following criteria to evaluate the data:

- Was the data reported correctly (e.g., analysis of solids reported incorrectly as liquid, transcription errors, and unit conversion errors)?
- Was an analytical result actually a "less than" value (below detection limits)?
- Is the sample material still present in the tank?

The complete notes from the review are documented in WHC-SD-WM-CN-014, *Notes from WHC Review of Sample Records for TWRS Accelerated Safety Analysis*.

3.2 DEFINE WASTE GROUPS

Tanks and waste are grouped into waste models (Figure 2, step 2), which accomplishes several objectives:

- Increases the number of data points for each radionuclide concentration selection
- Accounts for tanks with no measured data

- Simplifies calculations
- Reduces error potential (fewer numbers to keep track of)
- Provides future operational flexibility (no restrictions on pumping between tanks in the same group)
- Increases conservatism when combined with the method used for selecting isotope concentrations.

Logical groups include the type of tank (SSTs versus DSTs), physical form of the waste (liquid versus solid), and type of waste (aging waste versus other waste). Previous accident analyses resulted in a need for an "all tank solids" and an "all tank liquids" group. The physical basis for the eight groups derived from this logic are shown in Figure 7. The eight groups are a realistic lower limit for the number of groups that may be used.

The SSTs primarily contain wet salt cake that has been through an evaporative process and/or sludge. Although these materials are solids with interstitial liquid, small volumes of supernatant liquid are sometimes present on top of the salt cake or sludge. Accidents in these tanks will primarily release solids, with some liquid.

The DSTs have large liquid volumes. The volume of solids in DSTs is relatively small except a few tanks that contain substantial solids layers. The liquids are of two major types: (1) concentrated products such as double-shell slurry, double-shell slurry feed, and complexant concentrate that are mainly from SSTs and that have been evaporated to a minimum practical volume; and (2) dilute waste waiting to be evaporated comes from SSTs stabilization, customer wastes, and clean out of chemical processing plants. The solids are carryover from the pumping process and solids that have precipitated out of solution returned from an evaporator. Accidents in these tanks primarily will release liquids with some solids.

The AWF tanks have large liquid volumes. The wastes in these tanks include wastes from the last PUREX runs. They have some of the highest radionuclide concentrations. The wastes in these tanks are mostly liquid and have not been through an evaporator.

Process knowledge-based assessment of the potential for more groups does not identify clear reasons for further subdivision. Further subdivision on the basis of measured concentrations, without a process basis, invalidates the assumption that the large number of data points improves representativeness of actual waste variations within and between tanks.

3.3 DEVELOP AND VALIDATE DATA

The process for data selection and validation is illustrated Figure 8. This figure is an expansion of process block 3 from Figure 2. All available data was collected, and the process followed several steps to complete the database and validate all data points.

Figure 7. Eight Waste Models Used for Grouping Tanks and Waste.

AWF = aging waste facility
 DST = double-shell tank
 SST = single-shell tank

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Figure 8. Development of Data Set Organized Data and Verified Points Selected for Unit Liter Doses.

WHC-SD-WM-ER-400, 1995, *Tank Waste Source Term Inventory Validation*, Rev 0.
Volumes I and II, Westinghouse Hanford Company, Richland, Washington.

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3.3.1 Selection of Radionuclide Activity Concentration

The concentrations of the radionuclides were selected by collecting and evaluating historical sample data and tank content estimates derived from flowsheet-based models and intertank transfer records. The data was compiled into electronic files by ICF KH. The database, containing more than 11,000 entries, is the largest single collection of usable sample data at the Hanford Site. This data was initially entered into a database without screening for duplicates or errors. The data was screened at a later date. The initial entry was intended to collect all available data. The data came from both the 100- and 200-series tanks. The data, procedures, and assumptions used to compile the database are given in WHC-SD-WM-ER-400. The data sources include the following:

- WHC-SD-WM-TI-543, *Radionuclide and Chemical Inventories for the Double-Shell Tanks*
- WHC-SD-WM-TI-565, *Radionuclide and Chemical Inventories for the Single-Shell Tanks*
- WHC-SD-WM-SARR-003, *High-Level Waste Tank Subcriticality Safety Assessment*
- Sample data collected by the WHC Tank Characterization Program (Tank Characterization Library-200 East)
- Miscellaneous Tank Characterization Reports
- WHC-SD-WM-TI-057, *TRAC: A Preliminary Estimation of the Waste Inventories in Hanford Tanks through 1980²*
- WHC-SD-WM-ER-349, *Historical Tank Content Estimate for the Northeast Quadrant of the Hanford 200 East Area²*
- WHC-SD-WM-ER-351, *Historical Tank Content Estimate for the Northwest Quadrant of the Hanford 200 West Area²*
- WHC-SD-WM-ER-350, *Historical Tank Content Estimate for the Southeast Quadrant of the Hanford 200 East Area²*
- WHC-SD-WM-ER-352, *Historical Tank Content Estimate for the Southwest Quadrant of the Hanford 200 West Area.²*

Except for the four radionuclides listed below, post-June 1989 data is available. For ULD calculation, the post-June 1989 data has been used, unless pre-June 1989 data is a larger concentration. The listed radionuclides are all low contributors to dose. An order of magnitude increase in the

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²Does not contain actual sample results; calculated concentrations of radionuclides and chemicals were derived from process flowsheets and historical data on transfers in and out of the Hanford Site tanks.

concentration of any of the four would have little effect on ULDs (Tables 4 and 5).

- ^{60}Co in AWF liquid
- ^{154}Eu in AWF liquid
- ^{238}Pu in AWF liquid
- ^{241}Pu in DST liquid, DST solids and AWF solids.

3.3.2 Analysis of Sample Information

Analysts generated scatter plots (see WHC-SD-WM-ER-400) for each of 11 selected radionuclides, with liquid and solid samples plotted on separate plots, using tank waste characterization data from the sources previously identified. This process was repeated for six of the waste groups. Note that two models (one for all solids and one for all liquids) do not introduce new data; they present the highest data for each nuclide from the corresponding three liquid or solid models, respectively. Activity concentrations were plotted on the vertical axis, and tank numbers were plotted on the horizontal axis.

Each plot has a related table that lists all the sample data for that plot from highest value to lowest value. The tables contain tank identification, sample number, sample date, type of sample, sample results as reported in the original laboratory analysis, sample results converted to activity concentrations (Bq/L), sample results decayed to December 1994, and comments. The tables are updated for the TWRS FSAR by adding the sample data taken during 1995. The 1995 sample data and the high (bounding) points selected from pre-1995 data are decayed to September 1996.

Subject matter experts, who are familiar with the history and processes at the Hanford Site, evaluated the plots. The team had representatives from the analytical laboratory, process chemistry, waste tank operations, TWRS engineering, ICF KH, and safety analysis. The process was to determine whether the largest value on the plot was a technically possible number that did not conflict with known process parameters or contain obvious errors. If the sample met these criteria, it was accepted and became the FSAR maximum sample activity concentration. If the sample did not meet these criteria, it was discarded and the group evaluated the next highest sample result. This process continued until a sample result was found that met the criteria. Usually, the evaluation was made using the original sample results reported by the analytical laboratory or a process engineering support group. Only if neither of these sources were available, were derivative reports that referenced a prior source used. If no direct link to the analytical laboratory could be established, a reported sample result was not used.

This analysis includes both the 100- and 200-series SSTs. All the high points came from the 100-series SSTs. The process history for the tanks indicates that the wastes in the tanks should not be significantly different, except possibly for tanks C-201 through C-204. Because tanks C-201 through C-204 received waste from the strontium semi-works pilot plant, some experts believe these tanks could contain high strontium concentrations. Existing sample data does not support this view; however, analyses of future samples from these tanks for strontium would help clarify this point.

Table 4. Percent Contribution of Major Radionuclides to the Total Unit Liter Dose Based on GENII* Ingestion, Ground Shine, and Resuspension Calculation. (2 sheets)

Isotope	Bq/L	ULD Sv/L	% of ULD	Bq/L	ULD Sv/L	% of ULD
Single-shell tank liquids			Single-shell tank solids			
⁶⁰ Co	9.53 E+06	4.60 E-06	0.0	4.2 E+08	2.0 E-04	0.0
⁹⁰ Sr	1.05 E+10	2.50 E-02	48.1	1.6 E+12	3.6 E+00	87.8
⁹⁰ Y	1.05 E+10	2.20 E-03	4.2	1.6 E+12	3.2 E-01	7.8
¹³⁷ Cs	2.21 E+10	2.00 E-02	38.5	1.0 E+11	9.0 E-02	2.2
¹⁵⁴ Eu	2.35 E+09	4.40 E-04	0.8	5.8 E+09	1.0 E-03	0.0
²³⁷ Np	0.00	0.00	0.0	3.0 E+07	3.2 E-03	0.1
²³⁸ Pu	9.21 E+04	5.60 E-06	0.0	1.9 E+08	1.2 E-02	0.3
²³⁹ Pu	3.62 E+07	2.40 E-03	4.6	4.4 E+08	2.9 E-02	0.7
²⁴¹ Pu	2.57 E+08	3.30 E-04	0.6	3.2 E+09	4.1 E-03	0.1
²⁴¹ Am	4.23 E+07	2.90 E-03	5.6	2.3 E+08	1.6 E-02	0.4
²⁴⁴ Cm	4.23 E+05	1.40 E-05	0.0	2.3 E+06	8.8 E-05	0.0
Total ULD		5.20 E-02	102.49		4.1 E+00	99.11
Double-shell tank liquids			Double-shell tank solids			
⁶⁰ Co	6.97 E+06	3.40 E-06	0.0	1.5 E+07	7.2 E-06	0.0
⁹⁰ Sr	4.59 E+09	1.00 E-02	14.7	5.2 E+10	1.2 E-01	25.0
⁹⁰ Y	4.59 E+09	9.10 E-04	1.3	5.2 E+10	1.0 E-02	2.1
¹³⁷ Cs	5.86 E+10	5.30 E-02	77.9	5.9 E+10	5.3 E-02	11.0
¹⁵⁴ Eu	4.18 E+07	7.50 E-06	0.0	3.0 E+08	5.5 E-05	0.0
²³⁷ Np	2.30 E+05	2.30 E-05	0.0	8.1 E+05	8.1 E-05	0.0
²³⁸ Pu	1.78 E+06	1.10 E-04	0.2	7.2 E+07	4.4 E-03	0.9
²³⁹ Pu	7.65 E+06	5.20 E-04	0.8	1.6 E+09	1.1 E-01	23.1
²⁴¹ Pu	1.84 E+07	2.30 E-05	0.0	3.8 E+09	4.8 E-03	1.0
²⁴¹ Am	3.40 E+07	2.30 E-03	3.4	2.7 E+09	1.9 E-01	39.6
²⁴⁴ Cm	1.22 E+05	4.50 E-06	0.0	1.6 E+07	6.0 E-04	0.1
Total ULD		6.80 E-02	98.38		4.8 E-01	102.90

Table 4. Percent Contribution of Major Radionuclides to the Total Unit Liter Dose Based on GENII* Ingestion, Ground Shine, and Resuspension Calculation. (2 sheets)

Isotope	Bq/L	ULD Sv/L	% of ULD	Bq/L	ULD Sv/L	% of ULD
	Aging waste facility liquids			Aging waste facility solids		
⁶⁰ Co	7.71 E+05	3.70 E-07	0.0	4.9 E+08	2.4 E-04	0.0
⁹⁰ Sr	5.60 E+09	1.20 E-02	13.0	2.9 E+12	6.5 E+00	80.2
⁹⁰ Y	5.60 E+09	1.10 E-03	1.2	2.9 E+12	5.8 E-01	7.2
¹³⁷ Cs	8.84 E+10	7.80 E-02	84.8	9.8 E+10	8.7 E-02	1.1
¹⁵⁴ Eu	0.00	0.00	0.0	1.1 E+10	2.0 E-03	0.0
²³⁷ Np	9.20 E+04	9.20 E-06	0.0	9.9 E+08	9.9 E-02	1.2
²³⁸ Pu	2.75 E+03	1.70 E-07	0.0	6.7 E+07	4.1 E-03	0.1
²³⁹ Pu	1.20 E+06	8.10 E-05	0.1	4.4 E+08	2.9 E-02	0.4
²⁴¹ Pu	3.39 E+05	4.30 E-07	0.0	1.7 E+09	2.1 E-03	0.0
²⁴¹ Am	1.10 E+06	7.60 E-05	0.1	1.1 E+10	7.6 E-01	9.4
²⁴⁴ Cm	1.10 E+04	4.10 E-07	0.0	6.1 E+07	2.4 E-03	0.0
Total ULD		9.20 E-02	99.20		8.1 E+00	99.58
	All liquids			All solids		
⁶⁰ Co	9.53 E+06	4.60 E-06	0.0	4.9 E+08	2.4 E-04	0.0
⁹⁰ Sr	1.05 E+10	2.50 E-02	22.7	2.9 E+12	6.5 E+00	79.3
⁹⁰ Y	1.05 E+10	2.20 E-03	2.0	2.9 E+12	5.8 E-01	7.1
¹³⁷ Cs	8.84 E+10	7.80 E-02	70.9	9.8 E+10	8.7 E-02	1.1
¹⁵⁴ Eu	2.35 E+09	4.40 E-04	0.4	1.1 E+10	2.0 E-03	0.0
²³⁷ Np	2.30 E+05	2.30 E-05	0.0	9.9 E+08	9.9 E-02	1.2
²³⁸ Pu	1.78 E+06	1.10 E-04	0.1	1.9 E+08	1.2 E-02	0.1
²³⁹ Pu	3.62 E+07	2.40 E-03	2.2	1.6 E+09	1.1 E-01	1.3
²⁴¹ Pu	2.57 E+08	3.30 E-04	0.3	3.8 E+09	4.8 E-03	0.1
²⁴¹ Am	4.23 E+07	2.90 E-03	2.6	1.1 E+10	7.6 E-01	9.3
²⁴⁴ Cm	4.23 E+05	1.40 E-05	0.0	6.1 E+07	2.3 E-03	0.0
Total ULD		1.10 E-01	101.29		8.2 E+00	99.48

*PNL-6584, 1988, GENII - The Hanford Environmental Radiation Dosimetry Software System, Vol. 1, Pacific Northwest Laboratory, Richland, Washington.
ULD = unit liter dose.

Table 5. Percent Contribution of Major Radionuclides to the Total Unit Liter Dose Due to Inhalation. (2 sheets)

Isotope	Bq/L	ULD Sv/L	% dose	Bq/L	ULD Sv/L	% dose
	Single-shell tank liquids			Single-shell tank solids		
⁶⁰ Co	9.53 E+06	5.63 E-01	0.0	4.2 E+08	2.5 E+01	0.0
⁹⁰ Sr	1.05 E+10	6.79 E+02	6.2	1.6 E+12	1.1 E+05	47.7
⁹⁰ Y	1.05 E+10	2.39 E+01	0.2	1.6 E+12	3.7 E+03	1.7
¹³⁷ Cs	2.21 E+10	1.91 E+02	1.7	1.0 E+11	8.7 E+02	0.4
¹⁵⁴ Eu	2.35 E+09	1.82 E+02	1.7	5.8 E+09	4.4 E+02	0.2
²³⁷ Np	0.00	0.00	0.0	3.0 E+07	4.4 E+03	2.0
²³⁸ Pu	9.21 E+04	9.76 E+00	0.1	1.9 E+08	2.0 E+04	9.1
²³⁹ Pu	3.62 E+07	4.20 E+03	38.3	4.4 E+08	5.1 E+04	23.1
²⁴¹ Pu	2.57 E+08	5.73 E+02	5.2	3.2 E+09	7.2 E+03	3.2
²⁴¹ Am	4.23 E+07	5.08 E+03	46.3	2.3 E+08	2.7 E+04	12.5
²⁴⁴ Cm	4.23 E+05	2.83 E+01	0.3	2.3 E+06	1.5 E+02	0.1
Total ULD		1.10 E+04	100.00		2.2 E+05	99.99
	Double-shell tank liquids			Double-shell tank solids		
⁶⁰ Co	6.97 E+06	4.12 E-01	0.0	1.5 E+07	8.6 E-01	0.0
⁹⁰ Sr	4.59 E+09	3.11 E+02	4.9	5.2 E+10	3.3 E+03	0.6
⁹⁰ Y	4.59 E+09	1.09 E+01	0.2	5.2 E+10	1.2 E+02	0.0
¹³⁷ Cs	5.86 E+10	5.06 E+02	8.4	5.9 E+10	5.1 E+02	0.1
¹⁵⁴ Eu	4.18 E+07	3.23 E+00	0.1	3.0 E+08	2.3 E+01	0.0
²³⁷ Np	2.30 E+05	3.36 E+01	0.6	8.1 E+05	1.2 E+02	0.0
²³⁸ Pu	1.78 E+06	1.89 E+02	3.1	7.2 E+07	7.6 E+03	1.4
²³⁹ Pu	7.65 E+06	8.82 E+02	14.7	1.6 E+09	1.8 E+05	34.5
²⁴¹ Pu	1.84 E+07	4.10 E+01	0.7	3.8 E+09	8.5 E+03	1.6
²⁴¹ Am	3.40 E+07	4.08 E+03	67.4	2.7 E+09	3.3 E+05	61.5
²⁴⁴ Cm	1.22 E+05	8.17 E+00	0.1	1.6 E+07	1.1 E+03	0.2
Total ULD		6.06 E+03	99.99		5.3 E+05	100.00

Table 5. Percent Contribution of Major Radionuclides to the Total Unit Liter Dose Due to Inhalation. (2 sheets)

Isotope	Bq/L	ULD Sv/L	% dose	Bq/L	ULD Sv/L	% dose
	Aging waste facility liquids			Aging waste facility solids		
⁶⁰ Co	7.71 E+05	4.56 E-02	0.0	4.9 E+08	2.9 E+01	0.0
⁹⁰ Sr	5.60 E+09	3.62 E+02	25.4	2.9 E+12	1.9 E+05	10.8
⁹⁰ Y	5.60 E+09	1.28 E+01	0.9	2.9 E+12	6.6 E+03	0.4
¹³⁷ Cs	8.84 E+10	7.63 E+02	53.6	9.8 E+10	8.5 E+02	0.0
¹⁵⁴ Eu	0.00	0.00	0.0	1.1 E+10	8.6 E+02	0.0
²³⁷ Np	9.20 E+04	1.34 E+01	0.9	9.9 E+08	1.4 E+05	8.4
²³⁸ Pu	2.75 E+03	2.92 E-01	0.0	6.7 E+07	7.1 E+03	0.4
²³⁹ Pu	1.20 E+06	1.39 E+02	9.8	4.4 E+08	5.1 E+04	3.0
²⁴¹ Pu	3.39 E+05	7.56 E-01	0.1	1.7 E+09	3.7 E+03	0.2
²⁴¹ Am	1.10 E+06	1.32 E+02	9.3	1.1 E+10	1.3 E+06	76.5
²⁴⁴ Cm	1.10 E+04	7.37 E-01	0.1	6.1 E+07	4.1 E+03	0.2
Total ULD		1.42 E+03	100.00		1.7 E+06	100.00
	All liquids			All solids		
⁶⁰ Co	9.53 E+06	5.63 E-01	0.0	4.9 E+08	2.9 E+01	0.0
⁹⁰ Sr	1.05 E+10	6.79 E+02	5.8	2.9 E+12	1.9 E+05	9.9
⁹⁰ Y	1.05 E+10	2.39 E+01	0.2	2.9 E+12	6.6 E+03	0.4
¹³⁷ Cs	8.84 E+10	7.63 E+02	6.5	9.8 E+10	8.5 E+02	0.0
¹⁵⁴ Eu	2.35 E+09	1.82 E+02	1.5	1.1 E+10	8.5 E+02	0.0
²³⁷ Np	2.30 E+05	3.36 E+01	0.3	9.9 E+08	1.4 E+05	7.7
²³⁸ Pu	1.78 E+06	1.89 E+02	1.6	1.9 E+08	2.0 E+04	1.1
²³⁹ Pu	3.62 E+07	4.20 E+03	35.7	1.6 E+09	1.8 E+05	9.7
²⁴¹ Pu	2.57 E+08	5.73 E+02	4.9	3.8 E+09	8.5 E+03	0.5
²⁴¹ Am	4.23 E+07	5.08 E+03	43.2	1.1 E+10	1.3 E+06	70.4
²⁴⁴ Cm	4.23 E+05	2.83 E+01	0.2	6.1 E+07	4.1 E+03	0.2
Total ULD		1.17 E+04	99.99		1.9 E+06	99.98

ULD = unit liter dose.

In general terms, the criteria used by the evaluation group were as follows.

- Has the data been reported correctly (e.g., transcription errors, unit conversion errors, analyses of solids reported incorrectly as liquids)?
- Is an analytical result actually a less-than value (below detection limits)?
- Is the sampled material still present in the tank?
- Is the sample result compatible with known physical facts and process parameters?

The results of the review are documented in WHC-SD-WM-ER-400. In addition to the results of the review, the document contains a summary of assumptions, reasons that sample results were rejected, details on data presentation, and a complete set of plots and tables. The notes made at the review meetings conducted by the WHC evaluation group are contained in WHC-SD-WM-CN-014.

Details of the selected concentration values for the four radioisotopes that are the major contributors to the 8 inhalation ULDs are presented in Tables 6 through 9. Six of the 8 waste models are shown in column 1 because the "all liquid" and "all solid" models use the highest of the applicable concentrations (liquid or solid) from these 6 models. The four radioisotopes presented are ^{241}Am , ^{137}Cs , $^{239/240}\text{Pu}$, and ^{90}Sr . The concentration of the selected data point is given in column 2. The decayed value of the data point that matches the concentration given in Table 5 is listed in column 3. The percent contribution of the radionuclide to the ULD is given in column 4. The ratio of the selected data point to the next lower data point in a different tank is given in column 5. This ratio indicates that while the value picked is the highest valid concentration in the database, there is usually another concentration almost as large in another tank. The number of tanks in each grouping from which sample data was available is given in column 6. Observations from the expert review team on the quality and conservatism of the selected data point are given in column 7. More detailed information is presented in Appendix C of this document.

3.3.2.1 Derived Radionuclide Values. Sample data was not available for some radionuclides for some waste groups, and data for some radionuclides was not available for any group.

Activity concentrations for these cases were developed using known relationships between radioisotopes. The isotopes and waste groups for which calculated or implied values were used are listed in Table 10.

The laboratories do not routinely measure for ^{90}Y . Therefore, the values for ^{90}Y are based on its equilibrium relationship to ^{90}Sr , for which sample data is available. ^{90}Sr and ^{90}Y exist in an equal activity relationship. Therefore, the reported activity concentration of ^{90}Sr is equal to the ^{90}Y activity concentration.

Table 6. Evaluation of ^{241}Am Concentrations Specified for Waste Models.

Waste model	Selected data point (Bq/L)	Decayed data point 9/96*	% of inhalation ULD	Ratio to next lower point	Data availability	Confirmation information
SST liquid	3.7 E+07 1988 sample from 241-A-102	4.23 E+07	46.3	1.3	71% (106/149) of the SSTs	- Russian data solubility limit is 4.9 E+07 Bq/L - DST liquids model is 3.4 E+07
SST solid	1.42 E+08 1988 sample from 241-A-102	2.29 E+08	12.5	2.5	83% (124/149) of the SSTs	Samples from 3 different tanks within 40% of selected value
DST liquid	3.4 E+07 1985 sample from tank 241-AN-107	3.40 E+07	67	4	79% (19/24) of non-AWF DSTs	^{241}Am more soluble in highly complexed waste. - All values from non-organic tanks two orders of magnitude lower
DST solid	2.7 E+09 1988 sample from tank 241-SY-102	2.71 E+09	62	10^2	42% (10/24)	- Tank 241-SY-102 received ^{241}Am separations waste from PFP - Ten highest concentrations from 241-SY-102 - Almost two orders of magnitude higher than any other tank
AWF liquid	1.1 E+06 1985 sample from 241-AY-101	1.10 E+06	9	2.3	100% (4/4)	- Only AWF tank with organic complexant is 241-AY-101 - Solubility of ^{241}Am in organic higher than in aqueous - Homogeneous liquid
AWF solid	1.1 E+10 1989 sample from 241-AZ-101	1.10 E+10	77	2	100% (4/4)	- Tanks 241-AZ-101 and -102 contain waste from last PUREX campaign. - Airlift circulators mixed the waste - homogeneous.

* Section 3.3 explains how the original sample data was adjusted to account for the decay of ^{241}Pu to ^{241}Am . Adjusted values are found in Table 11. This column gives the decayed value of the Table 5 concentration. Thus, some decayed values are larger than the original sample data.

AWF = aging waste facility.
DST = double-shell tank.
PFP = Plutonium Finishing Plant
PUREX = Plutonium-Uranium Extraction (facility)
SST = single-shell tank.
ULD = unit liter dose.

Table 7. Evaluation of ^{137}Cs Concentrations Specified for Waste Models.

Waste model	Data point (Bq/L)	Decayed data point 9/96	% of inhalation ULD	Ratio to next point	Data availability	Confirmation information
SST liquid	2.3 E+10 1989 sample from tank 241-A-102	2.21 E+10	1.7	1.05	74% (110/149) of SSTs	No specific tank or group of tanks has been identified as being most likely to have highest concentration
SST solid	1.1 E+11 1974 sample from tank 241-SX-102	1.01 E+11	0.4	1.35	56% (83/149) of SSTs	- Tank was receiving evaporator bottoms and REDOX ion exchange waste. This could have contributed to high ^{137}Cs concentration - This value bounds concentrations reported for ferrocyanide tanks
DST liquid	6.1 E+10 1985 sample from tank 241-SY-103	5.86 E+10	8.4	1.25	92% (22/24) of DSTs	Cs soluble in aqueous. Highest concentration expected in highest concentrated evaporator returns, which includes 241-SY-103.
DST solid	6.1 E+10 1985 sample from tank 241-SY-103	5.86 E+10	0.1	1.07	66% (16/24) of DSTs	- Cs soluble in aqueous, therefore should be from interstitial liquid. - Four of five high samples from DSTs.
AWF liquid	9.2 E+10 1989 sample from tank 241-AZ-101	8.84 E+10	53.6	1.2	100% (4/4) of the AWF tanks	- Should be in tanks containing last PUREX run, which includes this tank. - Large liquid volume should be homogenous.
AWF solid	1.0 E+11 1990 sample from tank 241-AZ-101	9.80 E+10	0.0	1.07	100% (4/4) of the AWF tanks	Should be in tanks containing last PUREX run, which includes this tank.

AWF = aging waste facility.
 DST = double-shell tank.
 PUREX = Plutonium-Uranium Extraction (facility)
 REDOX = reduction oxidation.
 SST = single-shell tank.
 ULD = unit liter doses.

Table 8. Evaluation of $^{239/240}\text{Pu}$ Concentrations Specified for Waste Models.

Waste model	Data point (Bq/L)	Decayed data point 9/96	% of inhalation ULD	Ratio to next point	Data availability	Confirmation information
SST liquid	3.6 E+07 1986 sample from tank 241-C-106	3.6 E+07	38.3	1.2	68% (102/149) of SSTs	WSRC-TR-93-056, <u>Solubility of Plutonium and Uranium in Alkaline Salt Solutions</u>,[*] gives solubility limit as 0.0017 g/L. This sample concentration converts to 0.0157 g/L, ten times the WSRC-TR-93-056 limit
SST solid	4.4 E+08 1975 sample from tank 241-SX-115	4.4 E+08	23.1	1.1	51% (76/149) of SSTs	- Multiplication of this concentration times total SST sludge volume yields ten times total Pu sent to SSTs - Expected tank because it received high-level boiling waste from REDOX, and it was allowed to self-concentrate - Small solid volume (12,000 gal)
DST liquid	7.7 E+06 1988 sample from tank 241-AW-103	7.7 E+06	14.7	1.3	79% (19/24) of DSTs.	- Multiplication of this concentration times all DST liquid is over twice total Pu sent to DSTs - Bulk of Pu <u>should</u> be in solids
DST solid	1.6 E+09 1989 sample from tank 241-SY-102	1.6 E+09	34.5	10	50% (12/24) of DSTs	- Process knowledge supports max concentration in tank 241-SY-102 - Tank farm Pu tracking data shows this tank as highest - Calculated tank Pu = 187.3 kg compared to total of 83 kg sent to <u>all</u> DSTs
AWF liquid	1.2 E+06 1989 sample from 241-AZ-102	1.2 E+06	9.8	2.7	100% (4/4) of the AWF tanks	- Latest (most efficient) PUREX runs, should have low Pu as it does - Large liquid volume should be homogenous
AWF solid	4.4 E+08 1989 sample from tank 241-AZ-101	4.4 E+08	3.0	1.6	75% (3/4) of the AWF tanks	- Airlift circulators operated when filling - should be homogenous - Total PUREX Pu estimated at 83 kg sent to 12 tanks. This tanks Pu conc calculates to 25.4 kg, very conservative

^{*}WSRC-TR-93-056, Solubility of Plutonium and Uranium in Alkaline Salt Solutions (U), 1993, Savannah River Technology Center, Westinghouse Savannah River Company, Aiken, South Carolina.

AWF = aging waste facility.

DST = double-shell tank.

PUREX = Plutonium-Uranium Extraction (facility)

REDOX = reduction oxidation.

SST = single-shell tank.

ULD = unit liter doses.

Table 9. Evaluation of ^{90}Sr Concentrations Specified for Waste Models.

Waste model	Data point (Bq/L)	Decayed data point 9/96	% of inhalation ULD	Ratio to next point	Data availability	Confirmation information
SST liquid	1.1 E+10 1977 sample from tank 241-C-106	1.05 E+10	6.2	1.02	81% (121/149) of the SSTs	241-C-106 was the primary receiver near the end of Sr recovery operations 241-C-106 is the only remaining SST high-heat tank
SST solid	1.7 E+12 1977 sample from tank 241-AX-104	1.63 E+12	47.7	1.05	93% (139/149) of the DSTs.	241-AX-104 data point review validated the data - PUREX sludge - S and SX farms also expected to have high Sr concentrations. Second highest sample was from S farm
DST liquid	4.8 E+09 1985 sample from tank 241-AN-107.	4.59 E+09	4.9	1.05	96% (23/24) of the DSTs.	- DST-AN-107 collected evaporator returns from strontium recovery waste - High concentration of organic complexants, which enhances strontium solubility
DST solid	5.2 E+10 1995 sample from tank 241-AW-102	5.16 E+10	0.6	1.4	50% (14/28) of the DSTs	- Process knowledge indicates DSTs should have less strontium than SSTs or AWF tanks - Small contribution to dose
AWF liquid	5.8 E+09 1985 sample from tank 241-AY-101	5.60 E+09	25.4	1.3	100% (4/4) of the AWF tanks	- Highest organic complexant concentration tank. Strontium more soluble in organic than aqueous - Received PUREX aging waste - Recent sample 8.9 E+07 Bq/L
AWF solid	3 E+12 1989 sample from tank 241-AZ-101	2.88 E+12	10.8	1.9	100% (4/4)	- Highest sample expected from this tank as it received waste from most recent, highest efficiency PUREX runs. Sr more concentrated - Airlift circulators used when filling, solids well mixed

AWF = aging waste facility.
 DST = double-shell tank.
 PUREX = Plutonium-Uranium Extraction (facility)
 SST = single-shell tank.
 ULD = unit liter doses.

Table 10. Implied Concentrations of Radionuclides.

Radionuclide	Waste group	Implied concentration Bq/L
⁹⁰ Y	Single-shell tank liquid	1.05 E+10
	Single-shell tank solids	1.63 E+12
	Double-shell tank liquid	4.59 E+09
	Double-shell tank solid	5.16 E+10
	Aging waste facility liquid	5.6 E+09
	Aging waste facility solid	2.88 E+12
²³⁸ Pu	Single-shell tank solid	1.87 E+08
	Double-shell tank liquid	1.78 E+06
²⁴¹ Pu	Single-shell tank liquid	2.57 E+08
	Single-shell tank solid	3.22 E+09
	Double-shell tank liquid	1.84 E+07
²⁴⁴ Cm	Single-shell tank solids	2.29 E+06
	Single-shell tank liquid	4.23 E+05
	All liquids	4.23 E+05
	Aging waste facility liquid	1.10 E+04

Plutonium isotopes other than ²³⁹Pu were produced during the ²³⁹Pu production missions. Plutonium isotopes are inferred from the ²³⁹Pu concentration for waste models with inadequate sample data. A mixture containing 12 wt% ²⁴⁰Pu is assumed. High ²⁴⁰Pu also contains high concentrations of other transuranic isotopes. Some plutonium was created at the Hanford Site with higher than 12 wt% ²⁴⁰Pu. However, using 12 wt% is conservative because most of the production at the Hanford Site was 6% ²⁴⁰Pu, (the more desirable concentration for the weapons production mission). The ORIGEN2 calculations show an average of about 5 wt% ²⁴⁰Pu in the tanks.

Limited sample data was available for ²⁴⁴Cm. After looking at several methods, it was determined that ²⁴⁴Cm concentrations would be derived from a ratio of ²⁴⁴Cm to ²⁴¹Am. The ²⁴⁴Cm to ²⁴¹Am ratio is also a function of time. The ²⁴¹Am is building up because of ²⁴¹Pu beta decay and simultaneously decaying because of its own activity. At the same time, the ²⁴⁴Cm is decaying also. A brief analysis that explains the ratio developed for ²⁴⁴Cm concentration is included as Appendix A. The recommended ratio is 1×10^{-2} . The inhalation dose per curie of ²⁴⁴Cm is only 56% of that for ²⁴¹Am. The dose due to ²⁴⁴Cm will therefore always be small (0.01×0.56) compared with the dose from ²⁴¹Am.

The ^{241}Am concentration may increase because it is a daughter product of ^{241}Pu and the ^{241}Am has a longer half life than the ^{241}Pu . The behavior of the ^{241}Am depends on time and the relative concentrations of the ^{241}Pu and ^{241}Am . The ^{241}Am concentration is given by the following equation:

$$A_a = \lambda_p / (\lambda_a - \lambda_p) A_{p0} [(\exp(-\lambda_p t) - \exp(-\lambda_a t))] + A_{a0} \exp(-\lambda_a t) \quad (1)$$

where:

A_a = ^{241}Am activity as a function of time (Bq)
 λ_p = ^{241}Pu decay constant = 0.048 yrs^{-1}
 λ_a = ^{241}Am decay constant = 0.0016 yrs^{-1}
 A_{p0} = initial ^{241}Pu activity (Bq)
 A_{a0} = initial ^{241}Am activity (Bq)
 t = time (yr).

The ^{241}Am activities for the next 40 years were calculated using equation 1 and displayed in Table 11.

The peak americium activity for each composite was used in the dose calculation. This approach is very conservative because it is possible that peak ^{241}Pu and peak ^{241}Am concentrations are located in different tanks. However, using the peak value assures that conservative doses will be calculated.

3.4 DEFINE CONCENTRATIONS AND ULDs

The ULDs and the contributions from each of the 11 major isotopes for the 8 waste composites comprise the sum of 24-hour ingestion, resuspension, and ground shine doses are listed in Table 4. These ULDs were developed using the GENII code (PNL-6584, *The Hanford Environmental Radiation Dosimetry Software System*) assuming a unit X/Q. The offsite receptor is assumed to ingest fruits and vegetables and to receive doses from ground shine and resuspension during a 24-hour period. Twenty-four hours is assumed to be the time required to implement emergency response actions to limit further doses. In some cases the total may be more than 100% because of round off errors. GENII displays results to only two significant figures. The dominant isotopes for all composites are ^{90}Sr , ^{90}Y , ^{137}Cs , ^{239}Pu , and ^{241}Am .

The 8 ULD compositions, the activity concentration of each of the 11 major dose contributors, and the contribution of each radionuclide to the inhalation ULDs are listed in Table 5. The table may be used to assess the sensitivity of the ULD values to a change in the activity concentration for a particular radionuclide. For example, a 10% change in the ^{241}Am activity concentration for SST liquids results in a 4.63% change ($46.3\% \times 0.1$) in the SST liquid ULD. The four radionuclides discussed in the balance of this section account for over 95% of the dose (see Table 5).

Table 11. Americium Activity as Function of Time.

Mix	Initial ^{241}Pu activity (Bq/L)	Initial ^{241}Am activity (Bq/L)	^{241}Am at t = 10 yr	^{241}Am at t = 20 yr	^{241}Am at t = 30 yr	^{241}Am at t = 40 yr
SST liquid	2.80 E+08	3.70 E+07	3.99 E+07	4.15 E+07	4.22 E+07	4.23 E+07
SST solids	3.50 E+09	1.42 E+08	1.51 E+08	2.08 E+08	2.22 E+08	2.29 E+08
DST liquid	2.00 E+07	3.40 E+07	3.37 E+07	3.33 E+07	3.29 E+07	3.24 E+07
DST solids	4.10 E+09	2.70 E+09	2.71 E+09	2.70 E+09	2.67 E+09	2.64 E+09
AWF liquid	3.70 E+05	1.10 E+06	1.09 E+06	1.07 E+06	1.06 E+06	1.04 E+06
AWF solids	1.80 E+09	1.10 E+10	1.08 E+10	1.07 E+10	1.05 E+10	1.04 E+10

AWF = aging waste facility.

DST = double-shell tank.

SST = single-shell tank.

3.5 ASSESS CONSERVATISM

Assessment of information validity and application must account for the end use. Safety analysis is performed to establish appropriate controls to protect the health and safety of the public. Information must be sufficient for this purpose. Additional detail may be interesting, but it is a waste of resources.

Most of the concentration values include some amount of waste blending, at least within the finite size of the sample. Because sample sizes are very small relative to the quantity of waste that would have to be released to significantly effect the public at the site boundary, this blending should still lead to overestimation of larger volumes of release.

It is known that some tanks have discrete layers of waste. It would not be conservative to consider the waste at an "average" value, if the top layer is most likely to be released, and if the top layer contains the highest concentrations. The method of selecting the highest radionuclide value is intended to account for this possibility. (NOTE: Previous analyses attempted to take credit for layering when the top layer was known to be of lower concentration than the tank average. This analysis does not take credit for layering.)

3.5.1 Validation

The total radionuclide inventory was verified using Hanford Site production data (ORIGEN2 code calculations) and the track radioactive components (TRAC) inventory tracking database (WHC-SD-WM-TI-057). These methods only provide information on the relationship of the concentrations used in developing ULDs to overall average tank concentration, not to an individual tank. Individual tanks are known to contain inventories much above the average. However, overall risk is more related to the average tank

contents than the peak tank contents, unless there is some reason tanks with higher concentrations have higher failure risk than average.

The TRAC program (WHC-SD-WM-TI-057) estimates radioactive inventories in the SSTs and is useful in that the tank inventory estimates were generated independent of the efforts described in Section 3.1. TRAC estimates were derived from process flowsheets. Radionuclide inventory data from ORIGEN2 (ORNL-5621) was used as a source of data to derive the radioactive inventory estimates from the TRAC model. TRAC contains information on 68 different radionuclides and models the pathway of these radionuclides from the fuel, through processing, and into the tanks.

Although TRAC (WHC-SD-WM-TI-057) provides average concentration data, it is still a useful source of information because it contains a large number of radionuclides based on ORIGEN2 modeling that begins with the fuel and factors in process information. As shown in Table 12, the 8 radionuclides that contribute about 99% of the total dose in TRAC are ^{90}Sr , ^{90}Y , ^{241}Pu , ^{137}Cs , ^{238}Pu , $^{239/240}\text{Pu}$, and ^{241}Am . These radionuclides are a subset of the major contributors selected for the FSAR. The remaining radionuclides considered for the FSAR (i.e., ^{60}Co , ^{154}Eu , ^{237}Np , and ^{244}Cm) either contribute <1% to the inhalation dose in TRAC or are radionuclides that are important for shielding considerations only.

Both ORIGEN2 and TRAC identify the same major radiological dose contributors. These models account for all of the main radionuclides expected to be in the fuel as well as in the tanks.

Table 12. TRAC^a Total Activity Inventory and Inhalation Dose Contributions.

Nuclide	TRAC total activity ^b (Ci)	Dose conversion factor ^c (rem/Ci)	TRAC dose (rem)	TRAC percent total dose
^{241}Am	4.3 E+04	4.4 E+08	1.9 E+13	43.01
^{90}Sr	5.7 E+07	2.0 E+05	1.2 E+13	26.09
^{239}Pu	2.1 E+04	4.3 E+08	9.1 E+12	20.48
^{240}Pu	4.9 E+03	4.3 E+08	2.1 E+12	4.78
^{137}Cs	3.0 E+07	3.0 E+04	9.0 E+11	2.03
^{90}Y	5.8 E+07	8.7 E+03	5.1 E+11	1.14
^{241}Pu	5.7 E+04	8.2 E+06	4.7 E+11	1.05
^{238}Pu	1.1 E+03	3.9 E+08	4.3 E+11	0.97

^a WHC-SD-WM-TI-057, 1984, TRAC: A Preliminary Estimation of the Waste Inventories in Hanford Tanks Through 1980, Rockwell Hanford Operations, Richland, Washington.

^b TRAC total activity includes only the single shell tanks. TRAC activities are decayed to the end of 1990.

^c Inhalation dose conversion factor for a 50-year dose commitment period, units of rem committed effective dose equivalent per curie inhaled.

3.5.2 Conservatism

The ratio of the total safety analysis activity to the total production-based inventory for each radionuclide is given in the next to last column in Table 13. The FSAR inventories are 7 to 1,473 times the ORIGEN2 inventories. In other words, the activity concentrations correspond to total tank inventories that are 7 to 1,473 times the total activity produced in the reactors at the Hanford Site for most of the major dose contributors. Dose weighted conservative values for each nuclide and their totals are shown in Table 13. Because the nuclides with the extreme values both high and low (^{137}Cs , ^{237}Np) are very small contributors, they have very little effect on the overall conservatism (Table 13). The average overall conservatism is 58, near the conservatism of the three highest ULD contributors.

The various contributors to conservatism in the concentrations and ULDs developed by this process are summarized in Figure 9. More information on each contributor is provided in the following discussion.

Table 13. The Bounding Concentrations Used in ULDs Applied to Total Tank Inventory Produce Fifty-eight Times the Dose Calculated from Total Hanford Site Production of Radionuclides (Less Product).

Radionuclide	Production based % of total dose	Cumulative % of total dose	Ratio of safety analysis concentration based dose to production-based dose	Dose weighted conservatism ^a
^{241}Am	35.2	35.2	43	15.1
^{239}Pu	29.6	64.8	54	16.0
^{90}Sr	27.1	91.9	68	18.4
^{137}Cs	3.3	95.2	7	0.23
^{241}Pu	1.9	97.1	91	1.73
^{238}Pu	1.5	98.6	289	4.33
^{90}Y	1.2	99.8	68	0.82
^{244}Cm	0.2	100	40	0.08
^{154}Eu	0.1	100.1	92	0.09
^{237}Np	0.1	100.2	1,473	1.47
^{60}Co	0.0	100.2	167	0.0
Total	100.2 ^b			58.3

^a Dose weighted conservatism = (production-based %) (ratio of doses).
^b Total exceeds 100% because of rounding.
 ULD = unit liter doses.

Figure 9. Analysis of Many Methods to Substantiate Bounding Nature of Concentrations.

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Conservative Method:

WHC applies conservative methods for determining and applying radionuclide concentrations and ULDs in safety analyses. The most important features of the conservative methods for determining the concentrations and ULDs as shown in Figure 9 include the following:

Tank grouping – Grouping tanks and waste into eight waste models increased the number of data points for each radionuclide, as compared to calculating a value for each tank. Most individual tanks do not have samples of all radionuclides, and many tanks have no valid samples. If real variations exist between the tanks in a group, this method, which applies the highest data in the group, is conservative.

Highest data point – Selection of the highest data point for the group, rather than choosing alternative representations of an extreme (e.g., average plus three standard deviations) generally provides a conservative value. The number and distribution of the data points in the set determine how conservative this method is.

Highest valid data by isotope – Extreme values were used for each radionuclide. This is conservative relative to calculation of the ULD for each tank and compared to selecting the tank with the highest ULD to use in the model.

All available data – The initial database was created with all available data sources minus only the high-end data that was inappropriate, had demonstrable errors, or did not have valid records.

Historical Knowledge:

Historical knowledge used in deriving conservative concentrations included databases and records as well as reviews by experts with extensive Hanford Site experience. The following are examples of how historical data was used.

Reactor production – The operational history of all the reactors at the Hanford Site was used to calculate the total inventory of radionuclides. All this material, except that removed as product, was assumed to be present in the tank waste. Actual losses exist in the form of disposed low-level waste and contamination in the reactors and separations facilities.

Separations data – Operational records and personal knowledge about what campaigns were run and which tanks received specific waste types were used to evaluate that the distribution of radionuclides. The resulting concentrations are consistent with this knowledge.

Waste tracking data – The TRAC model and waste management operational data extended the separations data knowledge to movement of waste in and among the tank farms. The resulting concentrations are consistent with this knowledge.

Tank histories – Tank fill histories (WHC-SD-WM-ER-349, WHC-SD-WM-ER-352, WHC-SD-WM-ER-351, and WHC-SD-WM-ER-350) were evaluated to ensure no anomalous data points were selected. This information helped confirm that individual tank concentrations should not exceed the model waste concentrations.

Sample data – The subject matter experts evaluated each candidate concentration data point to ensure applicability and data reporting accuracy. Only samples that were traceable to the original sample records were used. Data points with the errors were rejected.

Temperatures – Analysts used recent tank temperature data and a heat transfer model to calculate total heat production in each passively ventilated SST. This data confirmed that no tanks showed anomalous heat production (i.e., the tank heat production data helped confirm that individual tanks do not exceed the applicable waste model concentration). Overall heat production data confirmed that the overall inventory was conservative.

Scientific Knowledge:

Where feasible, scientific knowledge confirmed the accuracy and/or conservatism of the radionuclide concentrations.

Solubility – Available radionuclide solubility data was assessed to ensure the reasonableness of concentrations in liquids because solubility for some radionuclides is significantly different in aqueous and organic liquids.

Chemistry – Subject matter experts used chemical knowledge to ensure data consistency. This includes relative concentrations between liquid and solid models, and among the various waste forms.

Radiochemistry – The ORIGEN2 (ORNL-5621) computer code represents consensus knowledge of radiochemistry. Subject matter experts used this knowledge to check concentration data. Inference was used in lieu of actual data only when the inferred data led to higher concentration values.

Radionuclide transport – Radionuclide transport knowledge was used in the development of ULDs from concentrations of radionuclides. This knowledge came in the form of experienced analysts and industry standard computer codes.

Heat flow – Analysts used heat transfer knowledge to calculate total heat production in a tank. This data was compared to heat production calculated from waste model isotope concentrations, which confirmed that overall concentrations are very conservative, and that there are no anomalous individual tanks.

Statistical Knowledge:

Pacific Northwest National Laboratory (PNNL) evaluated the tank data from a statistical view point. The subject matter experts also employed simple statistical methods to derive and validate the individual sample concentrations selected for the waste models.

Scatter plots – The selection process began with the highest data point from a plot of each waste model's sample data concentration for each radionuclide plotted on a scatter diagram versus each tank. These plots demonstrate that in most cases, the vast majority of the data points lie well (orders of magnitude) below the selected data point. They also show that there is usually more than one data point within an order of magnitude of the selected point.

Probability distributions – PNNL performed a quantitative analysis of rejected data points and of one isotope to confirm that unevaluated data points were unlikely to contain data that would exceed the selected values.

PNNL also developed distributions for 24 groups (4 radioisotopes x 6 models) (see Appendix D). The "all liquids" and "all solids" waste models were not plotted because they contained the high points picked from the 6 waste models plotted. The tables in Appendix D identify the mean, median, standard deviation, and the 95% and 99% point, compared with the selected point (identified as "Source" in the Appendix D tables). These distributions demonstrate that the analyte point selection method is conservative, and can be used to compute a less conservative result from alternative strategies. Overall, Appendix D demonstrates that the points selected are close to the strategy of selecting the 99% point directly from the observed distribution. Appendix D also substantiates that the empirical linearly interpolated estimates of the 99% point are more reliable than those obtained by fitting a gamma distribution to the data. The reason is that the gamma distribution points are more determined by the assumed distribution than by the data. In many cases, for important nuclides, the gamma distribution at 99% exceeds the source term selected point, while for 20 out of 24 of the empirical 99% points, the source term point exceeds the statistical point. Finally, in the four cases in which the empirical point exceeds the source term selected point, the difference is small.

Measurement uncertainty – The subject matter experts considered measurement uncertainty in developing the strategy for waste models, and in evaluation of data. Uncertainty due to nonhomogeneous waste and uncertainty due to sampling limitations and methods far outweigh radioisotope analytical uncertainty. This knowledge led to the grouping of tanks to increase the data set size for each radionuclide, and to the preference for using derived concentrations when measured data were low. Both methods added to the overall conservatism.

3.5.3 Unit Heat Loads in Passively Ventilated SSTs

The heat loads for passively ventilated SSTs were estimated using average vapor space temperatures, as described in WHC-EP-0709, *Estimation of Heat Load in Waste Tanks Using Average Vapor Space Temperatures*. The tank-by-tank specific heat rates for the passively ventilated SSTs as a function of cumulative tank waste volume is shown in Figure 10 (WHC-EP-0182-81). Data points in the figure show how much of the total volume of SST waste possesses heat loads below a particular heat rate value.

Figure 10. Independent Measurement of Tank Heat Loads Using Thermocouple Data Shows that Using Radioisotope Activity Concentrations Produces More Conservative Heat Load.

*Tank unit heat loads calculated from vapor space temperatures

WHC-EP-0182-81, 1995, *Waste Tank Summary Report for Month Ending December 31, 1994*, Westinghouse Hanford Company, Richland, Washington.

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This method is primarily a check on the ^{137}Cs and ^{90}Sr concentrations, because these two isotopes are the primary heat producers.

The average liquid waste volume for all of the SSTs included in Figure 10 was 50%. A unit heat load was calculated based on a mixture of 50% SST liquids and 50% SST solids using the activity concentrations for the SST liquids and solids composite source terms. This unit heat load (0.161 W/L) is shown as a horizontal line on Figure 10. The unit heat load developed from the activity concentrations used in the FSAR bounds all of the SSTs except tank 241-A-105, which represents only 0.05% (19 kgal) of the total SST waste (see Figure 10).

A vast majority of the SSTs have unit heat loads that are an order of magnitude or more less than the FSAR value. This comparison is useful because it compares predicted heat loads to what can be considered measured heat loads, although it is recognized that the heat loads were inferred from vapor space temperatures. The heat load estimates do not depend on the specific contents of the tank, and therefore provide an independent means of demonstrating the validity of the FSAR source term activity concentrations.

3.5.4 Statistical Methods

PNNL performed a statistical analysis of the radiological concentrations. The complete report is included as Appendix B. The report addresses two main questions. (1) Is it possible that there were errors in the reporting of some smaller concentrations that caused them to appear smaller than the selected value? (2) What confidence can be attached to the selected value and how likely is it that the selected value is the highest value in the database?

A statistical sample of 75 ^{90}Sr concentrations in SST solids demonstrated that at least 96.1% of the reported ^{90}Sr values are smaller than the recommended concentrations with 95% confidence. The samples were selected from the WHC-SD-WM-ER-400 database. The types of errors were not specific to, or more or less likely for, particular radionuclides. The type of tank (SST, DST, or AWF) from which the sample was obtained did not affect the error rate either. Thus, the results for ^{90}Sr should apply to all of the radionuclides.

According to the report, the ^{90}Sr statistical conclusions can be extended to all 8 model tanks and all 11 radionuclides. This is because most of the errors observed are errors that could be made for any analyte, rather than errors that are specific to an individual analyte. In particular, the errors that can lead a reported value to have been lower than the selected ULD value when it should have been higher are usually errors in conversion or reporting of units of measurement.

PNNL also prepared distributions for the data on the 4 radioisotopes analyzed in Section 3.3 (see Appendix D).

3.5.5 Overall Conservatism

Reviews of this report questioned the method, asserting that the method was too conservative. Alternate choices could produce results that are less conservative, and may be appropriate after completion of the first iteration of the safety analysis. Alternative methods, the potential impacts, and the pros and cons for the alternative approaches are presented in Table 14.

Details on the distribution of data for the radionuclides that contribute the most to calculated dose are provided in Appendix D. This information can be used to make quantitative estimates of the impact of alternative strategies. For example, the ULDs developed with the present strategy are very close to directly selecting the 99 percentile from the empirical data distribution as shown in the tables in Appendix D.

There is not a compelling case for changing to an alternative method for selecting the ULDs (see Table 14). The many contributors to the overall conservative approach to the safety analysis are discussed in Section 3.5. A systematic evaluation of all of the methods is necessary to decide the appropriate level of conservatism required in selecting ULDs. The basis used should be evaluated after the current safety analysis activity is completed and revised as appropriate.

4.0 OTHER FACILITIES AND INTERFACES

4.1 INACTIVE MISCELLANEOUS UNDERGROUND STORAGE TANKS

IMUSTs were used for the following functions associated with plutonium and uranium separations and waste management activities in the 200 East and 200 West Areas:

- Settling solids before disposal of supernate in cribs or a reverse well
- Neutralizing acidic process wastes before disposal in a crib
- Receiving and processing of SST waste for uranium recovery operations
- Collecting water that intruded diversion boxes and transfer pipeline encasements and any leakage that occurred during waste transfer operations
- Storing hexone from S-Plant (reduction and oxidation [REDOX])
- Receiving flush and decontamination solutions
- Diverting flow streams
- Waste handling and process experimentation.

Table 14. Alternative Methods for Selection of Data Points
Influence Overall Conservatism.

Method	Potential impact	Pros	Cons
Base method (eight models, highest data point in model)	None; base model	Conservative Simple Maximizes use of existing sample data	Low reviewer acceptance Perceived as overly conservative
Eight models, highest actual tank represented by the model, highest ULD calculated for a single tank	>20% reduction in total ULD	Potentially more realistic. Accounts for variations from tank-to-tank	Some tanks have no sample data More calculations
Statistical data point (e.g., average plus three standard deviations)	Not assessed	Higher reviewer acceptance	Non-continuous (including, in some cases, apparent bimodal) statistical distributions do not support this approach
Other statistical choice, e.g., 97th percentile (third data point down of a sample of 100)	>20% reduction in total ULD	Potential for higher reviewer acceptance Eliminates individual high points that may be in error or represent very low probability concentration, but are procedurally correct	Less conservative
More models	Potentially substantial for limited numbers of tanks	Reduces unnecessary conservatism More realistic representation of risks Potential for more economic application of TSRs and other controls	Adds safety analysis calculations Introduces more uncertainty (fewer data points per sample population) Limited by current sample data

TSR = Technical Safety Requirement.
ULD = unit liter doses.

Thirty-six IMUSTs have been identified as part of TWRS facilities, and more are expected to be identified. Most of the IMUSTs contain some amount of waste and are interim isolated and interim stabilized. Interim isolated means the tanks are isolated from facilities by sealing all inlet and outlet connections to the tanks. Interim stabilized means that all liquid that is technically and economically practical to remove has been removed. The total remaining volume of liquids and solids in the IMUSTs is approximately 0.4% of the volume of the wastes in the SSTs (WHC-EP-0861).

A general ULD for IMUSTs was not developed because sample information is unavailable and the IMUSTs are used for different purposes. The tanks need to be evaluated on a case-by-case basis by taking samples from each tank and/or process history.

4.2 INTERFACING FACILITIES

4.2.1 204-AR Waste Unloading Facility

The 204-AR Waste Unloading Facility is a receiving station for radioactive wastes that are brought in railcars and tank trailers from Hanford Site facilities outside the tank farms. This facility provides the capability of pumping waste from railcars and tank trailers to the high-level waste tanks. The waste is normally pumped to the DSTs.

The *Safety Analysis Report for the 204-AR Waste Unloading Facility* (WHC-SD-WM-SAR-040) does not specify either a maximum dose rate or maximum concentrations for radionuclides that will be accepted at the facility. Instead, WHC-SD-WM-SAR-040 refers to the maximum concentrations that can be shipped as defined in the *Safety Analysis Report for Packaging Railroad Liquid Waste Tank Cars* (WHC-SD-RE-SAP-013). Tank trailer shipments are governed by similar requirements found in WHC-SD-TP-SEP-011, *Safety Evaluation for Packaging for the 222-S Laboratory Cargo Tank*, and applicable sections of 49 CFR 173.425, "Transport Requirements for Low Specific Activity (LSA) Radioactive Materials." Agreements (internal memos) between the generators and the 204-AR Waste Unloading Facility specify maximum concentrations of radionuclides that will be shipped. The concentrations, for the radionuclides measured, are bounded by the concentrations used to calculate the liquid ULD for DST tanks.

It is recommended that the 204-AR Waste Unloading Facility adopt the same list of radionuclides and bounding concentrations that are used for the DST ULDs. This will ensure that waste transferred to tank farms is bounded by the tank farms' ULD.

4.2.2 242-A Evaporator

The 242-A Evaporator reduces the volume of liquid in the DSTs by evaporating the water in the waste. The 242-A Evaporator is operated in accordance with WHC-SD-WM-SAR-023, *242-A Evaporator/Crystallizer Safety Analysis Report*.

WHC-SD-WM-SAR-023 specifies radionuclide loading limits. The same limits apply to both feed and to slurry from the Evaporator. This prevents reconcentrating waste that has already been through the Evaporator. The limits are controlled by operating safety requirements (OSR). The specified radionuclide loading and the calculation of a ULD for the 242-A Evaporator are shown in Table 15.

The 242-A Evaporator ULDs are compared with six tank farm ULDs in Table 16. The output from the 242-A Evaporator is a slurry with an estimated 30% solids. Solids in the evaporator slurry will settle out in a tank, adding to the solids layer. The slurry from the 242-A Evaporator is sent to DSTs. The ULD of the slurry is slightly higher (7.7×10^3 Sv/L versus 6.1×10^3 Sv/L) than the ULD of DST liquid (see Table 13). It is almost two orders of magnitude smaller (7.7×10^3 Sv/L versus 5.3×10^5 Sv/L) than the ULD for DST solids. A slurry ULD for DSTs, composed of 70% DST liquid and 30% DST solids, would be approximately two orders of magnitude larger than the ULD for evaporator slurry.

Operation of the 242-A Evaporator and its discharge to the tank farms will not conflict with the ULDs developed for the tank farms. The 242-A Evaporator controls its operation to discharge limits that have a smaller ULD than the DSTs or the AWF tanks.

4.2.3 Transfer Lines

Any pumping operation out of a high-level waste tank or double-contained receiver tank may include a solids carry over. The amount of solids carry over is determined by the plant engineering organization. The ULD for transfers can be calculated by using the following sum:

ULD for transfers =

(Carry-over fraction) (Solids ULD) + (1 - Carry-over fraction) (Liquids ULD)

Addition of waste to SSTs was stopped in 1980. Remaining supernatant and interstitial liquid in the SSTs is pumped to DST's. Therefore, transfers from SSTs to DSTs can use the SST ULDs.

Transfers between and among DSTs and AWF tanks will make it increasingly difficult to identify the liquids or solids as being specifically DST or AWF source terms. For these transfers, the composite ULDs should be used. The composite ULDs were calculated by assuming the highest concentration of a radionuclide that is found in either a DST, an SST, or an AWF tank.

4.2.4 PUREX and the Plutonium Finishing Plant

The Criticality Prevention Specification for Tank Farms Operations (CPS-T-149-00010) specifies an investigation level for characterization samples for ^{239/240}Pu of 1 g/L in the settled solids. The limit for transfers to tank farms is expressed as 0.013 g/L of solid-liquid mixtures. Once the transfer is complete, solids with a concentration of 1 g/L could settle out of the mixture and form a solids layer on the top of the receiver tank.

Table 15. Radionuclide Loading Limits and 242-A Evaporator Unit-Liter-Dose Calculation.

Radionuclide	Source strength ($\mu\text{Ci/mL}$)	Source strength (Bq/L)	Dose conversion factor (Sv/Bq)	Unit liter dose (Sv/L) inhaled
^{14}C	2.6 E-01	9.6 E+06	5.6 E-10	5.4 E-03
^{60}Co	1.2 E+00	4.4 E+07	5.9 E-08	2.6 E+00
^{79}Se	7.8 E-02	2.9 E+06	2.7 E-09	7.7 E-03
^{90}Sr	2.2 E+02	8.1 E+09	6.5 E-08	5.3 E+02
$^{90}\text{Y}^*$	2.2 E+02	8.1 E+09	2.3 E-09	1.9 E+01
^{94}Nb	9.8 E-02	3.6 E+06	1.1 E-07	4.1 E-01
^{99}Tc	2.0 E+00	7.4 E+07	2.2 E-09	1.7 E-01
^{106}Ru	5.3 E+01	2.0 E+09	1.3 E-07	2.5 E+02
^{129}I	2.6 E-03	9.6 E+04	4.7 E-08	4.5 E-03
^{134}Cs	1.5 E+01	5.6 E+08	1.3 E-08	6.9 E+00
^{137}Cs	1.5 E+03	5.6 E+10	8.6 E-09	4.8 E+02
^{154}Eu	5.0 E+00	1.9 E+08	7.7 E-08	1.4 E+01
^{155}Eu	7.0 E+00	2.6 E+08	1.1 E-08	2.9 E+00
^{226}Ra	3.3 E-02	1.2 E+06	2.3 E-06	2.8 E+00
^{238}Pu	1.3 E-03	4.8 E+04	1.1 E-04	5.1 E+00
^{239}Pu	1.6 E-01	5.9 E+06	1.2 E-04	6.9 E+02
^{241}Pu	1.5 E+01	5.6 E+08	2.2 E-06	1.2 E+03
^{241}Am	1.0 E+00	3.7 E+07	1.2 E-04	4.4 E+03
^{244}Cm	1.3 E-02	4.8 E+05	6.7 E-05	3.2 E+01
Total				7.7 E+03

Source: WHC-SD-WM-SAR-023, 1995, 242-A Evaporator/Crystallizer Safety Analysis Report, Rev. 2, Westinghouse Hanford Company, Richland, Washington.

*While ^{90}Y is not in the WHC-SD-WM-SAR-023 list, this isotope is found in equilibrium with ^{90}Sr and was therefore included.

Table 16. Comparison Between Evaporator and Tank Farms Inhalation Unit Liter Doses.

Composite	Unit liter dose (Sv/L inhaled)
Evaporator	7.7 E+03
Single-shell tank liquids	1.1 E+04
Double-shell tank liquids	6.1 E+03
Aging waste facility liquids	1.4 E+03
Single-shell tank solids	2.2 E+05
Double-shell tank solids	5.3 E+05
Aging waste facility solids	1.7 E+06

The concentrations of $^{239/240}\text{Pu}$ used to calculate the ULDs for DST solids and AWF solids are at maximum 0.68 g/L and 0.19 g/L, respectively. No controls are in place that would prevent tank farms from accepting transfers from PUREX or the Plutonium Finishing Plant with plutonium concentrations that exceed the $^{239/240}\text{Pu}$ concentrations used to develop the ULDs for DST and AWF solids.

The concentration of $^{239/240}\text{Pu}$ in transfers to the tank farms should be consistent with the concentration of $^{239/240}\text{Pu}$ in the applicable ULD.

5.0 INFORMATION FOR TECHNICAL SAFETY REQUIREMENTS

Assumptions listed in this section are for use in the TWRS FSAR to identify TSRs or other safety controls.

1. Eight waste models provide a conservative and bias-free representation of the TWRS tanks and waste.
2. The sample data represents the radionuclide distribution within and among the tanks. The material is not assumed to be homogenous; rather, the number of samples acquired is large enough to represent the inhomogeneity.
3. Absent or unrealistically low sample data is better represented by implied concentrations derived from other samples.
4. Unsampld tanks are represented by the tanks that have been sampled.
5. For calculated isotope concentrations, decay is assumed to September 1996.

6. No credit is given for losses of radionuclides caused by holdup, removal from the stream by other than product (e.g., that which went out as low-level waste), or from tank leakage.
7. Ingestion of radionuclides following an accident can be interdicted within 24 hours.
8. Receptors will be subject to the same radionuclides and concentrations as determined for the waste model. That is, there are no chemical reactions or physical mechanisms (e.g., agglomeration, precipitation) that would change the mix.
9. Some selected sample data points are core composites. We recognize that the compositing process dilutes the sample concentration.

The following recommendations are presented:

1. The plutonium concentrations in settled solids received in wastes from other Hanford Site operations should not exceed the concentration of plutonium used in the solids ULD for the receiving tank.
2. The 204-AR Waste Unloading Facility should adopt the same list of radionuclides and bounding concentrations used for DST liquids to ensure that the dose from this facility does exceed the DST liquid ULD.
3. A TSR should be developed to ensure appropriate limits consistent with these ULDs.
4. An ongoing program should be maintained to review continuing sample analyses against the bounding concentrations used in the ULDs. In particular, concentrations of $^{89/90}\text{Sr}$ in the C Farm 200-series tanks and $^{239/240}\text{Pu}$ concentrations in TX-118 are of interest.

Supporting calculations (e.g., ORIGEN2 [ORNL-5621], tank heat load, TRAC [WHC-SD-WM-TI-057]) also include more detailed assumptions in their application. Because the method of selecting ULDs was conservative and there were multiple checks in the process, it is unlikely that assumptions in these support efforts would materially affect our conclusions.

6.0 CONCLUSIONS

The approach described in this report makes best use of available data and specifically meets the stated objectives as follow.

1. Radionuclide concentrations and ULDs were specified for use in the TWRS FSAR (see Table 2).

2. The scope included all 177 large waste storage tanks, the IMUSTs, and the interfacing lines and facilities (see Table 1). However, valid sample data was not available for creating radionuclide concentrations or ULDs appropriate to the IMUSTs. The analysis recommends ULDs for the 204-AR Waste Unloading Facility and for the 242-A Evaporator interfaces to the tanks. The current criticality specification permits shipment to the tank farms of higher concentrations of $^{239/240}\text{Pu}$ than the concentrations used to develop the tank ULDs. This difference must be resolved during development of the TSRs.
3. The use of conservative values in developing radionuclide concentrations and ULDs was substantiated through conservative methods, statistical analyses, process knowledge, and operations history.
4. A logical basis was developed for grouping the 177 tanks into six waste models by physical waste phase, waste characteristics, and tank physical characteristics; and two models that address all-solid and all-liquid phases.
5. Assumptions inherent in developing concentrations and ULDs were identified. The FSAR will consider these assumptions in developing TSRs to ensure the validity of the overall safety analysis process.
6. The majority of the accidents considered in the FSAR involve mechanical release mechanisms; thus the ULDs developed herein are directly applicable. For those accidents involving nonmechanical release mechanisms, appropriate modifications will need to be made to the ULDs.

These results were accomplished by deliberately identifying conservative radionuclide concentrations and ULDs that could be used in the safety analysis process. Overall conservatism was demonstrated by comparing overall production calculations and heat load data. To create a closer approximation to operational conditions and the different waste types, six different models were developed. The use of six models helped limit unnecessary conservatism while permitting bounding ULDs for different operations.

Evaluation of potential changes to the ULD process leads to a recommendation not to revise the process until all the conservative factors in the safety analysis process have been identified and evaluated. When the sensitivity of the entire safety analysis process for making changes in the ULD has been established, this report and its supporting database can be used to assess the need for additional characterization data to reduce unnecessary conservatism. If the overall process indicates that the calculations were too conservative, and if using high ULD values is shown to be the primary cause, alternative methods can be assessed in further detail.

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APPENDIX A

METHOD FOR ESTIMATING CURIUM-244 CONCENTRATION IN TANK FARM WASTE
WHEN SAMPLE DATA IS NOT AVAILABLE

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APPENDIX A

METHOD FOR ESTIMATING CURIUM-244 CONCENTRATION IN TANK FARM WASTE
WHEN SAMPLE DATA IS NOT AVAILABLE

The purpose of this appendix is to enable the estimation of ^{244}Cm concentration in specific waste tanks when sample data is not available. This will permit inclusion of estimated ^{244}Cm inventories in radiological dose calculations.

Curium-244 is produced by a series of (n, γ) reactions from either ^{241}Am or ^{241}Pu so that the maximum amount of ^{244}Cm is present at the end of irradiation (i.e., the maximum amount of ^{244}Cm does not increase during storage). The amount of ^{244}Cm in the tank waste is estimated by using available data to form ratios of ^{244}Cm activity to activities of isotopes closest to it in the production chain (i.e., ^{241}Am , ^{241}Pu , or $^{239/240}\text{Pu}$). Because of the removal of plutonium during fuel processing, one would expect that the ratios using the plutonium isotopes might be highly variable and not reliable as predictors of ^{244}Cm . A series of such ratios (^{244}Cm activity to activities of isotopes closest to it in the production chain) was calculated using the data in Table 1 in WHC-SD-WM-SARR-016, *Tank Waste Compositions and Atmospheric Dispersion Coefficients for Use in ASA Consequence Assessments*, which shows maximum sample activity concentrations for various sample groups. All the groups which showed a ^{244}Cm activity were used to generate Table A-1.

The ratio to ^{241}Am seems to be the most constant parameter covering a range of only about a factor of 6 (Table A-1, data column 1). The ratios using ^{241}Pu and $^{239/240}\text{Pu}$ (combined) are much more variable covering ranges of factors of about 80 and 40, respectively (Table A-1, data columns 2 and 3). The $^{244}\text{Cm}/^{241}\text{Am}$ activity ratio therefore seems to be the most reliable predictor of ^{244}Cm .

Table A-1. Sampled Isotope Ratios for Various Maximum Sample Groups (Ci/Ci).

Sample Group	$\frac{^{244}\text{Cm}}{^{241}\text{Am}}$	$\frac{^{244}\text{Cm}}{^{241}\text{Pu}}$	$\frac{^{244}\text{Cm}}{^{239/240}\text{Pu}}$
All liquids	3.5 E-03	4.6 E-04	3.6 E-03
All solids	5.9 E-03	1.6 E-02	4.1 E-02
Double-shell tank liquids	3.8 E-03	6.5 E-03	1.7 E-02
Double-shell tank solids	6.1 E-03	4.0 E-03	1.0 E-02
Aging waste facility solids	5.9 E-03	3.6 E-02	1.5 E-01
Double-shell tank flammable gas watch list	1.0 E-03	1.0 E-03	7.7 E-03

The most current and reliable analytical predictions of ^{244}Cm are contained in WHC-SD-WM-RPT-164, *Inventories for Low-Level Waste Tank Waste*. Results from WHC-SD-WM-RPT-164 are based on the detailed operating history of the Hanford Site using the ORIGEN2 computer code (ORNL-5621). The resulting composite tank composition, decay corrected to 1990 (WHC-SD-WM-SARR-016, Appendix H), yields a $^{244}\text{Cm}/^{241}\text{Am}$ activity ratio of 7.8×10^{-3} . Based on this result and the results from sample data shown in Table A-1, a conservative value of the $^{244}\text{Cm}/^{241}\text{Am}$ ratio equal to 1×10^{-2} is recommended.

It should be recognized that the $^{244}\text{Cm}/^{241}\text{Am}$ activity ratio is a function of time. The ^{241}Am is building up due to ^{241}Pu B decay and decaying due to its own activity. At the same time the ^{244}Cm is also decaying. ^{241}Pu has a half-life of 14.4 y ($\lambda_1 = 4.81 \times 10^{-2}/\text{y}$), ^{241}Am has a half-life of 432.2 y ($\lambda_2 = 1.6 \times 10^{-3}/\text{y}$), and ^{244}Cm has a half-life of 18.11 y ($\lambda_3 = 3.83 \times 10^{-2}/\text{y}$). Since the half-life of ^{241}Am is much longer than that of ^{241}Pu , the decay of the ^{241}Am can be neglected over the time scale of interest here, thereby simplifying the development of the following equations. The buildup of ^{241}Am can then be described by

$$(Am-241) = (Am-241)_0 + (Pu-241)_0 \frac{\lambda_2}{\lambda_1} [1 - \exp(-\lambda_1 t)]$$

where the isotope symbols refer to activities or activities per unit volume. The zero subscripts refer to initial activities at $t = 0$. Similarly the decay of the ^{244}Cm can be described by

$$(Cm-244) = (Cm-244)_0 \exp(-\lambda_3 t).$$

It is here assumed that, at time $t = 0$ (date the samples were decayed to), the initial activities of the ^{244}Cm and the ^{241}Am are related by an initial ratio, R_0 , equal to $(Cm-244)_0/(Am-241)_0$. It is then evident that the ratio ^{244}Cm to ^{241}Am , R , as a function of time is given by

$$R = R_0 \frac{\exp(-\lambda_3 t)}{1 + \frac{(Pu-241)_0 \lambda_2}{(Am-241)_0 \lambda_1} [1 - \exp(-\lambda_1 t)]}.$$

The numerator is a correction for decay of the ^{244}Cm and is constantly decreasing. The denominator is a corrector for buildup of ^{241}Am due to ^{241}Pu decay and is constantly increasing, but is significantly different from 1 only in the case of a very high ^{241}Pu to ^{241}Am ratio. Using the same data as used in Table A-1, the ^{241}Pu to ^{241}Am ratio is usually in the range 0.1 to 10. Values of R out to a time $t = 40$ y over this range of ^{241}Pu to ^{241}Am ratio are shown in Table A-2.

Table A-2. Calculated Values of $R(t)$, $R_0 = 0.01$.

Time (y)	$(^{241}\text{Pu})_0 / (^{241}\text{Am})_0$		
	0.1	1.0	10.
0	1.0 E-02	1.0 E-02	1.0 E-02
10	6.8 E-03	6.7 E-03	6.1 E-03
20	4.6 E-03	4.6 E-03	3.9 E-03
30	3.2 E-03	3.1 E-03	2.5 E-03
40	2.2 E-03	2.1 E-03	1.7 E-03

It is evident from these results that the amount of ^{244}Cm is constantly decreasing relative to ^{241}Am , but at a rate slow enough that a constant ratio of 0.01 will be usable for estimating ^{244}Cm for some time. The inhalation dose per Ci of ^{244}Cm is only about 56% of that for ^{241}Am . The dose due to ^{244}Cm will therefore always be small (<1%) compared with the dose from ^{241}Am .

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APPENDIX B
STATISTICAL ASSESSMENT OF RADIOLOGICAL SOURCE TERMS

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APPENDIX B

STATISTICAL ASSESSMENT OF RADIOLOGICAL SOURCE TERMS

B1.0 OVERVIEW

A statistical analysis of the radiological concentrations was performed for use in the calculation of accident consequences for WHC-SD-WM-SAR-067, *Tank Waste Remediation System (TWRS) Final Safety Analysis Report (FSAR)*. The radiological concentrations were obtained by examining the highest concentrations observed for each tank model from a database of tank sample analyses. The database was developed from the sources described in Section 3.3. The highest reported concentrations for each tank model were evaluated to determine if the concentrations were correct and appropriate for use in a unit liter dose (ULD). The value used for the ULD was the largest reported concentration that could not be rejected as inappropriate or erroneous.

Two main issues were addressed in the statistical analysis.

1. Is it possible that there are errors in the reporting of some of the smaller tank model concentrations that caused them to appear smaller than the selected value?
2. What confidence can be attached to the selected value - how likely is it that a higher concentration value than that used for the safety analysis is present in the database?

The statistical analysis concluded that at least 96.1% of the reported single-shell tank (SST) solid ^{90}Sr concentrations are smaller than the selected value, with 95% confidence. This statement can be applied to all radionuclides and all high-level waste storage tanks (under certain assumptions).

B2.0 ESTIMATION OF ERROR RATES

A statistical approach was used to determine if some of the reported tank concentrations not evaluated in the selection of the safety analysis values contained errors that would cause them to be larger than the selected value. Multiplicative errors of order 10^3 were deemed possible, but errors larger than that were deemed highly improbable. The multiplicative errors were assumed to be caused by errors in reporting or converting units of measurement. The analysis only addressed errors that would cause the reported concentration to be smaller than the correct concentration.

A statistical sample of 75 concentrations was randomly selected from those values larger than 10^{-3} times the selected value. The sample was constrained so that each data source (e.g., raw data, Braun [WHC-SD-WM-SARR-003, *High-Level Waste Tank Subcriticality Safety Assessment*], or tank characterization database [TCD] [Section 3.3]) was represented and so that no duplicate records of the same tank sample were included. The proportion of values from a particular source in the statistical sample was approximately equal to the proportion of values from that source in the entire database. From this database, a statistical sample of 75 concentrations was produced for ^{90}Sr in SST solids. Any radionuclide used in any ULD could have been used for the statistical sampling. ^{90}Sr was picked because it is a major contributor and data is plentiful.

B2.1 ANALYSIS OF THE STATISTICAL SAMPLE FOR ^{90}Sr

Seventy-five ^{90}Sr concentrations were randomly selected and examined for errors that caused the reported concentration to differ from what it should have been. Of the 75 values, 6 had errors. None of the errors were such that the correct concentration was larger than the concentration used in the SST solids ULD. A 95% confidence upper bound on the probability of any entry having an error that would cause it to be larger than the concentration used in the SST solids ULD is 3.9%. That is, even though no errors of this type were assumed, it is still plausible that any given entry in the database could have a 3.9% chance of having an error that would cause the entry to actually be larger than the concentration used in the SST solids ULD.

The only error that was larger than 2 orders of magnitude was in the negative direction (the actual concentration was smaller than reported).

B3.0 EXTENSION OF RESULTS

To justify the extension of the results for ^{90}Sr to the other radionuclides and to all high-level storage tanks, the reasons for rejecting data points in the original evaluation of the database (Section 3.3) were reviewed. Many concentrations were rejected using the process described in Section 3.3 not because of an error but because they were inappropriate. (Inappropriate means that the sample is no longer relevant because of continuing process activities in the tank farms.) The rejected data was divided into "errors" and "inappropriate data." The rejections classified as errors were identified according to whether they were the kind of error that could be made for any radionuclide (labeled "general" errors) or whether the error was something that would only be seen for a specific radionuclide (labeled "specific" errors). Results of the error-type count are listed in Table B-1.

Table B-1. Count of Error Types Observed.

		SST Solids	SST Liquid	DST Solids	DST Liquid	AWF Solids	AWF Liquid	Row Total
Am	Total	1	5	0	2	0	3	11
	Generic	0	1	0	0	0	0	1
	Specific	0	0	0	0	0	0	0
Cs	Total	15	21	1	2	1	1	41
	Generic	5	1	1	0	1	0	8
	Specific	0	0	0	1	0	0	1
Pu	Total	17	11	3	3	0	5	39
	Generic	4	3	3	1	0	0	11
	Specific	4	0	0	0	0	2	6
Sr	Total	9	7	1	1	0	7	25
	Generic	1	3	0	1	0	0	5
	Specific	0	0	1	0	0	0	1
Other	Total	5	20	1	0	0	2	28
	Generic	0	1	0	0	0	0	1
	Specific	1	0	1	0	0	0	2
Column Total	Total	47	64	6	8	1	18	144
	Generic	10	9	4	2	1	0	26
	Specific	5	0	2	1	0	2	10

AWF = aging waste facility.

DST = double-shell tank.

SST = single-shell tank.

Most (75%) of the samples rejected in the process of arriving at the ULD concentrations (as described in Section 3.3) were discarded not because of errors but because they were inappropriate for the ULDs. Of the remaining samples, 18% of the discarded values had errors that could occur for any radionuclide and only 7% had errors that were specific to that radionuclide. Considering only the concentrations that were rejected because of an error, 72% of these errors were general in nature. Although it is clear that the bulk of the errors is general in nature, it remains to be determined if the errors specific to certain radionuclides are of the kind that would cause the reported concentration to be substantially smaller than it should have been.

This analysis shows that errors specific to a particular analyte or assay are not a major component of the set of overall errors in the database. In particular, for ^{90}Sr , only one error was found (in the process of arriving at the ULD concentrations) that was specific to ^{90}Sr . The remainder were errors that could occur for any radionuclide. The predominate error that could cause a reported value to be considerably different from the actual value is human error, whether it is in transcription, data entry, units of measurement conversion, or spillage of the sample. This was borne out by both the examination of the data rejected in the process of arriving at the ULD concentrations and by the examination of the 6 errors that were discovered in the review of the 75 sampled strontium values. Thus, it is reasonable to assume that the errors made in ^{90}Sr values which cause the reported result to

differ substantially from the actual value are comparable in frequency and magnitude to errors of this type made in the reporting of other radionuclides. This justifies extending the estimate of 96.1% coverage to all of the radionuclides.

B3.1 STATISTICAL DESCRIPTION OF STRONTIUM VALUES

A statistical description of the behavior of the strontium values in the sampling frame is given in this section. Figure B-1 is a histogram of the strontium values.

The mean value is 1.51×10^{11} , the median is 1.73×10^{10} , and the standard deviation is 2.90×10^{11} . This is a rather large variance, indicating that the mean and median are not typical of many of the possible values. Also, the shape of the distribution is not that of a normal distribution. A gamma or a scaled beta distribution would provide a reasonable approximation.

B4.0 CONCLUSION

The statistical evaluation of randomly sampled ^{90}Sr values revealed no concentration values larger than the ULD value. The upper bound on the probability that any other ^{90}Sr value is larger than the selected value is 3.9%. That is, at least 96.1% of the ^{90}Sr reported concentrations are smaller than the ^{90}Sr ULDs. This statement is made with 95% confidence.

The 96.1% probability for ^{90}Sr can be extended to all radionuclides and to all high-level storage tanks. This is because most of the errors observed are errors that could be made for any analyte, rather than errors that are specific to an individual radionuclide. In particular, the errors that can lead a reported concentration to have been lower than the selected value, when it should have been higher, are usually errors in conversion or reporting of units of measurement.

B5.0 TECHNICAL APPENDICES

B5.1 SAMPLING PLAN

The sampling plan for ^{90}Sr was defined to be all laboratory analyses for SST solids that had produced results with the following qualities:

- No less than 10^{-3} X the source term value
- Not from track radioactive components (TRAC) (WHC-SD-WM-TI-057, *TRAC: A Preliminary Estimation of the Waste Inventories in Hanford Tanks Through 1980*) or Historical Tank Content Estimate (HTCE)

Figure B-1. Histogram of the Strontium Values.

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SRP-RAD

(WHC-SD-WM-ER-349, *Historical Tank Content Estimate for the Northeast Quadrant of the Hanford 200 East Area*; WHC-SD-WM-ER-350, *Historical Tank Content Estimate for the Southeast Quadrant of the Hanford 200 East Area*; WHC-SD-WM-ER-351, *Historical Tank Content Estimate for the Northwest Quadrant of the Hanford 200 West Area*; WHC-SD-WM-ER-352, *Historical Tank Content Estimate for the Southwest Quadrant of the Hanford 200 West Area*) sources

- Were not marked as "DUPLICATE" or "DUPLICATE RESULT" in sample state field
- If several results with the same sample identifier were reported, only one result was used.

It should be noted that because the sampling plan for ^{90}Sr excludes numerous observations with concentrations lower than the ULD concentration by a factor of 10^3 , the results given above are conservative because data less than 10^3 is excluded. If all the data were shown, the error rate would be lower. For ^{90}Sr , 48% of the values in the database were not included in the sampling plan. In order to extend the results given in the report from the laboratory data for ^{90}Sr to all tanks, one needs to assume that the top half (roughly) of the laboratory analyses is representative of the population of tanks.

Next observations were selected using a random selection process. The process included a feature that presented selecting duplicate samples. Of the 296 observations in the ^{90}Sr sampling plan, 118 were raw (taken from original lab reports), 25 were from Wastren (WHC-SD-WM-TI-565, *Radionuclide and Chemical Inventories for the Double-Shell Tanks*), 107 were from Braun (WHC-SD-WM-SARR-003), 36 were from the TCD, six were listed as "U110 - Core Data" taken from the tank characterization report (WHC-EP-0643, *Tank Characterization Report Single-Shell Tank 241-U-110*), and four were listed as "B110 - Core Data" taken from the tank characterization report (WHC-SD-WM-ER-368, *Tank Characterization Report for Single-Shell Tank 241-B-110*). The sample of 75 values was selected to be 30 raw data, six from Wastren, 27 from Braun, nine from the TCD, 2 from the "U110 - Core Data," and 1 from the "B110 - Core Data," thus preserving the relative proportions.

B5.1.1 Strontium Computer Code

The following code was used to randomly select the strontium samples used in this evaluation.

```
stron <- read.table("stront.csv", header=TRUE, sep=',',
  row.names = as.character(1:862), as.is=TRUE)
src <- unique(stron[,10])

#strip out the HTCE inventory estimates

stron <- stron[(stron[,10] != "HTCE Inv."),]
```

#Now drop everything with values smaller than ASA value by a factor
#of 10 to the third. ASA value is row 9, the one with the star in
#column 14.

```
stron <- stron[ (stron[,13] > stron[9,13]/1000) , ]
```

#And also drop the ones that have already been examined. That
#is, all of the ones up to and including the ASA value

```
stron <- stron[-(1:9), ]
```

#Now remove all the known duplicates, these have either "DUPLICATE" or
#"DUPLICATE_RESULT" in the sample_state field (column 9)

```
stron <- stron[ (stron[,9] != "DUPLICATE") ,]  
stron <- stron[ (stron[,9] != "DUPLICATE_RESULT") ,]
```

#Now remove the remainder of the duplicates

```
dupes <- NULL  
for (i in 1:(length(stron[,1])-1)) {  
  if (stron[i,4] != "") {  
    dupes <- c(dupes,  
              ((i+1):length(stron[,1]))[stron[-(1:i),4] ==stron[i,4]])  
  }  
}
```

```
stron <- stron[-dupes,]
```

```
src <- unique(stron[,10])
```

```
freq <- NULL  
for (i in 1:length(src)){  
  freq <- c(freq, sum(stron[,10] == src[i])/length(stron[,10]))  
}
```

#Number to take from each source given by

75*freq

#round it to

```
nsrc <- c(27,30,6,9,2,1)
```

#Generate sample:

```
save.seed <- .Random.seed
```

```
smpl <- NULL  
for (i in 1:length(nsrc)) {  
  blah <- stron[(stron[,10] == src[i]),]  
  n <- length(blah[,1])  
  foo <- sample(1:n,size = nsrc[i],replace=FALSE)  
  smpl <- rbind(smpl, blah[foo,])  
}
```

```
write(t(smpl[, -14]),file="stron.smpl",ncolumns=13,append=FALSE)
```

B5.2 CALCULATION OF CONFIDENCE LIMITS

The upper bounds on the probability that any entries in the database are in fact larger than the selected source term value were calculated as exact confidence limits based upon a binomial probability model.

Although a normal approximation to the binomial distribution is often sufficient for confidence intervals, this approximation was not used here. The normal approximation to the binomial model breaks down and yields inaccurate results when the probability of "success" (here, finding a value higher than the selected source term value) is very small. A general rule of thumb is that the observed number of "successes" is less than five, then the normal approximation is invalid. Here, the numbers observed were zero and one.

The exact confidence bound is calculated based on the first principle's definition of what a confidence bound is, and cannot be written as a closed-form expression. If the data X has a binomial distribution with n trials (here, $n = 75$) and "success" probability p (here $p =$ the probability that a higher result will be found), and the observed count is x , then the 95% upper bound is given by that value of p for which

$$0.05 = \sum_{y=0}^x P(y; n, p)$$

where $P(y; n, p)$ is the probability mass function of a binomial random variable with parameters n and p . For a description of this standard method of generating confidence intervals, (see Mood, Graybill, and Boes 1974, pages 392-393).

In practice, the confidence limit is solved by evaluating the above summation for a grid of values of p and choosing the p which gives a summation closest to 0.05. If greater accuracy is desired, the fineness of the grid can be enhanced. For $x=0$, in the strontium case, S-plus code to evaluate the above function for a grid of p values looks like this:

```
p <- seq(0,0.05,length=100)
prob <- dbinom(0,size=75,prob=p)
```

B6.0 REFERENCES

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APPENDIX C
DETAILED NOTES ON RADIONUCLIDE SELECTIONS

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APPENDIX C

DETAILED NOTES ON RADIONUCLIDE SELECTIONS

Detailed notes on selection of the concentrations of four major radionuclides are presented in this appendix. The concentrations are used in the calculation of unit liter doses for the *Tank Waste Remediation System (TWRS) Final Safety Analysis Report (FSAR)* (WHC-SD-WM-SAR-067). The notes reflect the judgments of the team that reviewed the waste tank sample data. The notes support the information given in the main text of this report, Tables 6 through 9.

²⁴¹Am. Single-Shell Tank Liquid - The data point used was 3.7×10^7 Bq/L,* from a 1988 sample taken from tank 241-AX-102.

The single-shell tanks contain liquid waste that is aqueous and high in hydroxide. Americium has a low solubility in aqueous waste. Work performed in the former Soviet Union indicates that the solubility of americium in a hydroxide solution, similar to that of the Hanford Site wastes, is 1.6×10^{-7} M (4.9×10^6 Bq/L) (WHC-EP-0817, *Transuranic Elements and Technetium and the Treatment of Alkaline Radioactive Wastes*). All of the waste tanks could have a concentration equal to the solubility limit in the liquid phase. Therefore, no particular tank has been identified as most likely to have a high concentration of americium. Instead, the tank with the highest concentration would be the result of americium being released to a tank when the temperature and pH in the tank are closest to ideal for keeping americium in solution. The historical database indicates that the highest valid sample was from tank 241-AX-102. The sample result, 3.7×10^7 Bq/L, is reasonably close to the theoretical maximum concentration, 4.9×10^7 Bq/L. In terms of curies, this is 1 mCi/L and 1.3 mCi/L, respectively. Additional evidence that solubility is the controlling factor for americium concentration in the liquid phase is that concentrations of americium in double-shell tank liquids and in the liquids from the flammable gas tanks subset are 3.4×10^7 and 3.3×10^7 Bq/L, respectively. Therefore, the point selected from tank 241-AX-102 is believed to be a good representation of the highest concentration that can be found in the single-shell tanks.

Sample data was available for 71% (106/149) of the single-shell tanks. The concentration used for the unit liter dose was 1.3 times greater than the next highest sample in the database. Because the concentration selected is very near the solubility limit, and more than 70% of the tanks have been sampled, it is considered very unlikely that a higher concentration will be found.

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*Bequerels/Liter is the appropriate metric unit for this report. Trying to convert this to M or other measures of concentrations by weight or volume may only cause confusion. In a few places a conversion was made to g/L to illustrate a point, in which case the Bq/L equivalent is in parenthesis. Activity concentrations in this appendix are decayed to December 1994. Data in the tables in the main document is further decayed to September 1996.

²⁴¹Am. Single-Shell Tank Solids - The data point used was 1.42×10^8 Bq/L from a 1988 sample taken from tank 241-A-102 of approximately 750 sample results entered in the database. Only nine were larger than 1×10^8 Bq/L, three of these were from the track radioactive components (TRAC) database, and were not used. The highest point, 4.23×10^8 Bq/L was taken from a laboratory report that stated that the analytical method used was still under development. The largest remaining valid point that was not a TRAC datapoint was from tank 241-A-102. Because of the drop in concentrations shown by the data (nine points above 1×10^8 Bq/L and more than 700 less than 1×10^8 Bq/L), the selected concentration is believed to be a good approximation of the highest concentration in single-shell tanks. Sample data for ²⁴¹Am was available from 83% (124 of 149) of the single-shell tanks.

²⁴¹Am. Double-Shell Tank Liquid - The data point used was 3.4×10^7 Bq/L, from a 1985 sample from tank 241-AN-107.

Some double-shell tanks have liquid waste containing organic complexants. Americium is more soluble in an organic complexant than it is in aqueous waste. Tank 241-AN-107 has the highest concentration of organic complexant, and it was logical to find it had the highest reported concentration of americium in the double-shell tanks. Of all the sample points in the historical database above the minimum plot line, all but one (from tank 241-AW-104) came from the organic complexant tanks 241-AN-107, 241-SY-101, and 241-AN-102. All values from nonorganic complexant tanks were about two orders of magnitude or more lower than the selected value.

Sample data was available for 79% (19/24) of the nonaging waste double-shell tanks; recent samples were available for the three complexant tanks. Because nearly 80% of the double-shell tanks have been sampled, and because of the preferential solubility of americium in organic complexant, it is very unlikely that a higher concentration will be found in a noncomplexant tank. A higher value may be reported from future analyses for the complexant tanks, but the value used for the unit liter dose was nearly four times greater than the next highest reported value.

²⁴¹Am Double-Shell Tank Solids - The data point used was 2.69×10^9 Bq/L, from a 1988 sample taken from tank 241-SY-102.

The highest concentration of americium in a double-shell tank would be expected to come from tank 241-SY-102 because it has been receiving americium separations wastes from the Plutonium Finishing Plant since 1976. The separations were done as part of a plutonium purification process. The other tanks that have high values of americium are two tanks containing cladding removal waste, the complexed concentrate tanks, and tank 241-SY-101, which has the most concentrated waste in the double-shell tank system.

Sample data was available for 42% (10/24) of the nonaging waste double-shell tanks. The low percentage of sampled tanks was not considered a problem because tank 241-SY-102 would have the highest concentration of americium of any double-shell tank. The ten highest concentrations reported in the database, representing four different sampling events, are from tank 241-SY-102. The concentration used for the unit liter dose was nearly two

orders of magnitude greater than the highest reported result from a tank other than tank 241-SY-102.

²⁴¹Am, Aging Waste Facility Liquid – The data point used was 1.1×10^6 Bq/L, from a 1985 sample taken from tank 241-AY-101.

The aging waste facility consists of four tanks, each of which holds a large volume of liquid. The tanks were sampled for liquids several times during the 1980's. As stated for single-shell tank and double-shell tank liquids, the solubility of americium in aqueous solutions is very low. Any tank could be at saturation for americium in the aqueous phase. Americium is more soluble in organic complexant than in aqueous solution. The only aging waste facility tank containing organic complexant is tank 241-AY-101, and therefore it should logically have the highest americium concentration of the aging waste facility tanks. The highest valid americium sample is from tank 241-AY-101. Because the samples were taken from large liquid volumes that were free to mix, the liquid was expected to be homogenous and dip samples to be representative of the tank. An additional sample taken late in 1994 from tank 241-AY-101 was not included in the historical database. The results for the 1994 sample were approximately 4.9×10^5 Bq/L, about one half the value used.

Sample data was available for 100% (4/4) of the aging waste facility tanks. The concentration used was 2.3 times greater than the highest reported value from any other aging waste facility tank. For the reasons discussed above, no tank should produce a concentration higher than what has been found in tank 241-AY-101. It is considered very unlikely that a higher concentration of americium will be found in the current contents of tank 241-AY-101 because the liquid is arguably homogenous and has been sampled on several occasions.

²⁴¹Am Aging Waste Facility Solids – The data point used was 1.1×10^{10} Bq/L, from a 1989 sample taken from tank 241-AZ-101.

Tanks 241-AZ-101 and 241-AZ-102 are filled with aging waste from the last Plutonium Uranium Extraction Facility (PUREX) campaign. Therefore, they should have the highest concentrations of americium of the aging waste facility tanks. There is no particular reason to say that one tank or the other would have a higher concentration of americium in the solids. When the tanks were filled, the airlift circulators were operating, and they continued operating for several years after the tanks were filled. The airlift circulators stirred the entire volume of liquid and solids. Therefore, the assumption is made that the solids were uniformly mixed when the circulators were running and that the 1989 sample is representative of the solids material in the tank.

Sample data was available for 100% (4/4) of the aging waste facility tanks. The concentration used for the unit liter dose was a factor of two higher than the next highest concentration reported in the database.

⁹⁰Sr Single-Shell Tank Liquid - The data point used was 1.1×10^{10} Bq/L, from a 1977 sample taken from tank 241-C-106.

⁹⁰Sr dissolved in liquid waste is the result of processing performed at B Plant to recover strontium from sludges. It is not a result of plutonium separations. The highest concentrations of strontium in liquid waste would be found in tanks that received waste from the strontium recovery operations. These wastes were discharged to several single-shell tanks during the 1970's. The wastes from these single-shell tanks were accumulated, shipped to the 200 West Area, evaporated, returned to the 200 East Area, and eventually collected in tank 241-AN-107, a double-shell tank. Tank 241-C-106 was the primary receiver near the end of the strontium recovery operations. Therefore, tank 241-C-106 should be a prime candidate for having the highest strontium concentration. Tank 241-C-106 is also the only remaining single-shell high-heat tank that requires water addition or forced ventilation flow to maintain waste temperature at safe levels. The heat is produced by radiolytic decay of several isotopes, of which strontium is a primary source. This also indicates that tank 241-C-106 should have a high, if not the highest, concentration of strontium in the single-shell tanks.

Sample data was available for 81% (121/149) of the single-shell tanks. The concentration used for the unit liter dose was 1.02 times larger than the next highest value highest value in the database.

⁹⁰Sr Single-Shell Tank Solids - The data point used was 1.7×10^{12} Bq/L, from a 1977 sample taken from tank 241-AX-104.

The highest strontium concentration should be found in aging waste that has not been through the strontium recovery process. All of the PUREX aging waste sludges have been recovered and processed to remove strontium except the material in the aging waste facility tanks. The Reduction and Oxidation (REDOX) Plant, T Plant, and B Plant also produced aging waste. The REDOX Plant waste would have the highest concentration of strontium because it was produced more recently (therefore undergoing less radioactive decay) and because the REDOX Plant process was more efficient than the B Plant or T Plant processes. The highest reported concentration was from tank 241-AX-104, which contains sludge from the PUREX. The second highest valid data point in the historical database was from tank 241-S-104, which received aging waste from the REDOX Plant and was not recovered for strontium removal. Because the tank 241-AX-104 data point was higher (enveloped) than the data point from tank 241-S-104, and the review group found no errors in the data, the tank 241-AX-104 data point was used although the S or SX tank farms were expected to have the high value.

Sample data was available for 93% (139/149) of the single-shell tanks. The concentration used for the unit liter dose was 1.05 times greater than the highest reported value from any other single-shell tank.

⁹⁰Sr Double-Shell Tank Liquid - The data point used was 4.79×10^9 Bq/L, from a 1985 sample taken from tank 241-AN-107.

As stated for strontium in single-shell tank liquid, the highest concentrations of strontium in liquid would result from the processing of

waste to recover it. The waste from strontium recovery was collected in single-shell tank receiver tanks, evaporated in the 200 West Area, returned to the 200 East Area, and collected in double-shell tank 241-AN-107. Also, tank 241-AN-107 has one of the highest concentrations of organic complexant in any of the tanks. The organic complexant would enhance strontium solubility. For these reasons, tank 241-AN-107 should be expected to show the highest concentration in the liquid phase, and the highest valid sample point in the historical database is indeed from tank 241-AN-107.

Sample data was available for 96% (23/24) of the nonaging waste double-shell tanks. The concentration used for the unit liter dose was 1.05 times greater than the highest reported value from any other double-shell tank.

⁹⁰Sr Double-Shell Tank Solids - The data point used was 5.2×10^{10} Bq/L, from a 1995 sample taken from tank 241-AW-102.

The aging waste facility solids should be expected to have the highest concentrations of strontium because they are from the last PUREX campaign and the strontium has not been recovered. However, the aging waste is in a separate category. Aging waste from B Plant, T Plant, and the REDOX Plant went to the single-shell tanks, not the double-shell tanks. PUREX aging waste went to single-shell tanks then to aging waste facility tanks, not double-shell tanks. The double-shell tanks should have lower strontium concentrations than single-shell tanks or aging waste facility tanks. In fact, the samples from the historical database show that there are approximately 10×10^{10} Bq/L for double-shell tanks versus approximately 10×10^{12} Bq/L for aging waste facility tanks. Because strontium only contributes 0.4% to the double-shell tank solids dose and strontium concentrations should stay below single-shell tank and aging waste facility tank levels, any variation in the unit liter dose would be small.

Strontium in the solid phase was not transferred from single-shell to double-shell tanks. Significant efforts were made not to transfer solids.

Sample data was available for 50% (14/28) of the double-shell tanks. The sample concentration used was 1.4 times greater than the next highest sample.

⁹⁰Sr Aging Waste Facility Liquid - The data point used was 5.84×10^9 Bq/L, from a 1985 sample taken from tank 241-AY-101.

Strontium is more soluble in organic complexant than it is in aqueous solution. Of the four aging waste facility tanks, tank 241-AY-101 has the highest organic complexant concentration. Because three of the tanks received aging waste from PUREX, the tank with the highest organic concentrate concentration that received PUREX aging waste would be expected to have the highest strontium concentration. The tank that meets these conditions is tank 241-AY-101, and it does have the highest valid sample point. The value, 5.84×10^9 Bq/L is very high, but there are no apparent reasons to discard the sample. A recent sample taken in accordance with the Waste Compatibility Data Quality Objectives criteria (WHC-SD-WM-DQO-001, *Data Quality Objectives for Tank Farms Waste Compatibility Program*) shows a concentration of

8.88×10^7 Bq/L. This sample was not included in the original database because of a timing factor. The recent sample indicates that the 1985 sample is very conservative.

Sample data was available from 100% (4/4) of the aging waste facility tanks. The concentration used was 1.3 times greater than the highest reported value from any other aging waste facility tank.

⁹⁰Sr Aging Waste Facility Solids - The data point used was 3×10^{12} Bq/L, from a 1989 sample taken from tank 241-AZ-101.

Either tank 241-AZ-101 or tank 241-AZ-102 would be expected to have the highest concentration of strontium in the solid phase because they contain primarily PUREX aging waste from the facility's last years of operation. PUREX was running at a higher efficiency when it was shipping waste to tank 241-AZ-101 than when tank 241-AZ-102 was being used. Therefore, using the high sample coming from tank 241-AZ-101 seems reasonable.

Although not the highest sample, many high samples come from tank 241-AY-102. Tank 241-AY-102 received transfers from both B Plant, where strontium separations were performed, and PUREX from 1980 through 1992. Analyses are not available, but some fission products, including strontium, were transferred to tank 241-AY-102. A core sample from tank 241-AY-102, taken in 1987, indicates high strontium but no significant quantities of ruthenium or cerium. This tends to support the idea that a strontium-rich stream from B Plant strontium separations, and not the aging waste facility, is responsible for the contents in tank 241-AY-102. The strontium separations waste would not have as high a concentration of strontium as nonseparated aging waste.

Sample data was available for 100% (4/4) of the aging waste facility tanks. The concentration used for the unit liter dose was 1.9 times greater than the highest reported value from any other aging waste facility tank. When the tanks were filled, the airlift circulators were operating, and they continued operating for several years after the tanks were filled. The airlift circulators stirred the entire volume of liquid and solids. Therefore, the assumption is made that the solids were uniformly mixed when the circulators were running and that the 1989 sample is representative of the solids material in the tank.

¹³⁷Cs in Single-Shell Tank Liquid - The data point used was 2.3×10^{10} Bq/L, from a 1989 sample taken from tank 241-A-102.

No specific tank, or group of tanks, has been identified that is expected to contain the largest concentration of cesium in the liquid phase. Because cesium contributes only 1.9% to the unit liter dose for single-shell tank liquid, a very large change in concentration would mean a small change in the unit liter dose.

Sample data was available for 74% (110/149) of the single-shell tanks. The concentration for the unit liter dose was 1.05 times greater than the next highest value.

¹³⁷Cs in Single-Shell Tank Solids - The data point used was 1.1×10^{11} Bq/L, from a 1974 sample taken from tank 241-SX-102.

As part of one of the early waste volume reduction campaigns at the Hanford Site, sodium ferrocyanide was added to some tanks. The ferrocyanide precipitated the cesium out of the liquid so that the liquid could be disposed of as low-level waste. Therefore, the highest concentrations of cesium would be expected in ferrocyanide watch list tanks. Several ferrocyanide watch list tanks yielded samples among the highest concentrations. However, a sample from 241-SX-102 has a concentration that is about 35% higher than any of the ferrocyanide tanks. The sample report appears valid. The tank 241-SX-102 datapoint was used.

Sample data was available for 56% (83/149) of the single-shell tanks. The concentration used for the unit liter dose was 1.35 times greater than the next highest value.

¹³⁷Cs in Double-Shell Tank Liquid - The data point used was 6.1×10^{10} Bq/L, from a 1985 sample taken from tank 241-SY-103.

Cesium is very soluble in aqueous liquid. The highest concentrations would be found in liquids that have been through one of the evaporators. The most concentrated aqueous liquid at the Hanford Site is double-shell slurry. Tanks 241-SY-101, 241-SY-103, and 241-AN-104 are the only tanks containing double-shell slurry, and the high sample came from tank 241-SY-103.

Sample data was available for 92% (22/24) nonaging waste double-shell tanks. The value used was 1.25 times greater than the highest reported value from any other double-shell tank.

¹³⁷Cs in Double-Shell Tank Solids - The data point used was 6.1×10^{10} Bq/L, from a 1985 sample taken from tank 241-SY-103.

The only major source of cesium in double-shell tank solids is interstitial liquid. The highest concentration of cesium in solids would be in the tanks with the most concentrated aqueous liquid, which are the double-shell slurry tanks 241-SY-101, 241-SY-103, and 241-AN-104. The high sample came from tank 241-SY-103. Four of the five high samples are from double-shell slurry tanks. The fifth is from tank 241-AW-102, which is an evaporator feed tank. The contribution to the unit liter dose from cesium is very small. A large change in cesium concentration would have little effect on the unit liter dose.

Sample data was available for 66% (16/24) of the double-shell tanks. The concentration used was 1.07 times greater than the next highest value.

¹³⁷Cs in Aging Waste Facility Liquid - The data point used was 9.2×10^{10} Bq/L, from a 1989 sample taken from tank 241-AZ-101.

The highest concentrations of ¹³⁷Cs should be in the last runs from PUREX. PUREX was the most efficient process and the last runs from PUREX were not processed to remove cesium. The aging waste from the last PUREX runs is in tanks 241-AZ-101 and 241-AZ-102, and the highest sample is from tank

241-AZ-101. Because the samples have been taken from large liquid volumes that are free to mix, the liquid is expected to be homogenous and dip samples are expected to be representative of the tank.

Sample data was available for 100% (4/4) of the aging waste facility tanks. The concentration used was 1.2 times greater than the highest reported value from any other aging waste facility tank.

¹³⁷Cs in Aging Waste Facility Solids – The data point used was 1.0×10^{11} Bq/L, from a 1990 sample taken from tank 241-AZ-101.

The highest concentrations of ¹³⁷Cs should be in the last runs from PUREX. PUREX was the most efficient process and the last runs from PUREX were not processed to remove cesium. The aging waste from the last PUREX runs is in tanks 241-AZ-101 and 241-AZ-102, and the highest sample is from tank 241-AZ-101.

It is somewhat anomalous that the ¹³⁷Cs concentration in the aging waste solids is slightly higher (about 8% higher) than in the aging waste liquid. The ¹³⁷Cs activity in the aging waste solids should be lower than the activity in the aging waste liquid. The discrepancy is within the analytical error limits traditionally associated with ¹³⁷Cs analyses.

Sample data was available for 100% (4/4) of the aging waste facility tanks. The concentration used was 1.07 times greater than the next highest value.

^{239/240}Pu in Single-Shell Tank Liquid – The data point used was 3.62×10^7 Bq/L, from a 1986 sample taken from tank 241-C-106.

In WSRC-TR-93-056, *Solubility of Plutonium and Uranium in Alkaline Salt Solutions (U)*, the upper limit of plutonium solubility in alkaline salt solutions representative of tank waste is reported as 0.0017 g/L. The reported concentration of plutonium in tank 241-C-106, converted to g/L, is 0.0157 (3.62×10^7 Bq/L), a factor of 10 greater than the upper limit given in WSRC-TR-93-056. This difference may be attributed to the presence of anions other than hydroxide or possibly organic complexant, all of which could result in higher concentrations of plutonium in the supernate. However, because of the factor of ten margin it was not investigated further.

Sample data was available for 68% (102/149) of the single-shell tanks. The concentration used was 1.2 times greater than the highest reported value from any other single-shell tank.

^{239/240}Pu in Single-Shell Tank Solids – The data point used was 4.4×10^8 Bq/L, from a 1975 sample taken from tank 241-SX-115.

Approximately 981 kg of plutonium were transferred to the tank farms, based on the recovery efficiency of the separations processes. Of this total, 83 kg were sent to double-shell tanks. Therefore, approximately 898 kg were sent to the single-shell tanks.

The highest concentration, based on sample results, is from tank 241-SX-115. This result was expected because tank 241-SX-115 received high-level, boiling waste from the REDOX Plant. The waste was allowed to self-concentrate, which means that the tank was allowed to remain at a high temperature so that liquid evaporated and the plutonium concentrated in the solids. The tank was declared an assumed leaker in 1965; the liquid was pumped out of the tank, but the solids were never sluiced from it. The total volume of solids in the tank is small, approximately 45,400 L (12,000 gal).

As a test of conservatism, the concentration, 0.1916 g/L (4.4×10^8 Bq/L), was multiplied by the total volume of sludge (46,045,748 L [12,164,000 gal]) reported for all the single-shell tanks. Sludge volume was used because the plutonium would be present in sludge, not in salt cake. The product of the concentration times the volume of sludge is 8,822 kg of plutonium. This quantity of plutonium is ten times the quantity of plutonium discharged to the single-shell tanks from the separations plants. It cannot be definitively stated that no future sample analysis will have a higher reported concentration of plutonium than tank 241-SX-115. However, it can be concluded that the concentration selected, 0.1916 g/L (4.4×10^8 Bq/L), overpredicts (and envelopes) the health consequences from plutonium in the single-shell tank solids unit liter dose.

The tank layer model in LA-UR-94-4269, *Tank Layer Model (TLM) Manual*, indicates that tank 241-TX-118 could have the highest plutonium concentration. This is not supported by the existing sample data. However, any future analyses of samples from 241-TX-118 should be checked for ^{239,240}Pu to ensure that the single-shell tank solid unit liter dose is using the high value.

Sample data was available for 51% (76/149) of the single-shell tanks. The concentration used was 1.1 times greater than the next highest value.

^{239,240}Pu in Double-Shell Tank Liquid – The data point used was 7.65×10^6 Bq/L, from a 1988 sample taken from tank 241-AW-103.

Approximately 981 kg of plutonium were transferred to the tank farms, based on the recovery efficiency of the separations processes. Of this total, 83 kg were sent to double-shell tanks.

The highest concentration, based on sample results, is from tank 241-AW-103. As a test of conservatism, the concentration from 241-AW-103, 2.9×10^{-3} g/L (7.65×10^6 Bq/L), was multiplied by the total volume of supernatant (60,683,963 L [16,031,000 gal]) reported for all the double-shell tanks. The product of the concentration times the volume of supernatant is 175 kg of plutonium. This quantity of plutonium is twice the quantity discharged to the double-shell tanks from the separations plants. This is a very conservative number because the bulk of the 83 kg of plutonium should be in the double-shell tank solids, not the liquid. Samples taken from the different tanks show concentrations in the solids range from 10 to 1,000 times greater than the concentration found in the liquid samples. The overprediction by a factor of two (175 kg versus 83 kg) combined with the ratio of plutonium in solids versus liquids (at least 10 to 1) indicates that the concentration used for the source term, 2.9×10^{-3} g/L (7.65×10^6 Bq/L), is conservative by at least a factor of 20. It cannot be definitively stated

that no future sample analysis will have a higher reported concentration of plutonium than that from tank 241-AW-103, especially because the upper limit of solubility of plutonium in an alkaline salt solution is 0.017 g/L. However, it can be concluded that the concentration selected, 2.9×10^{-3} g/L (7.65×10^6 Bq/L), overpredicts (and envelopes) the health consequences from plutonium in the double-shell tank liquid unit liter dose.

Sample data was available from 79% (19/24) double-shell tanks. The value used was 1.3 times greater than the highest reported value from any other nonaging waste double-shell tank.

^{239/240}Pu in Double-Shell Tank Solids - The data point used was 1.57×10^9 Bq/L, from a 1989 sample taken from tank 241-SY-102.

Approximately 981 kg of plutonium were transferred to the tank farms, based on the recovery efficiency of the separations processes. Of this total, 83 kg were sent to double-shell tanks. In addition, 23 kg of plutonium were transferred from the Plutonium Finishing Plant (PFP) to tank 241-SY-102. The waste from the PUREX Facility, which contained the 83 kg, was sent to approximately 12 tanks in the AP, AW, AY, and AZ tank farms. It is reasonable to assume that tank 241-SY-102, with 23 kg, has a higher concentration of plutonium in the solids than does any of the tanks in AP, AW, AY, or AZ tank farms. This idea is supported by the sample results. The highest reported concentration for tank 241-SY-102 is an order of magnitude greater than the next highest sample from a tank other than tank 241-SY-102 (tank 241-AW-103). The Tank Farm Plutonium Tracking System, a Hanford Site tracking system that uses tank transfer data sheets, also shows that the highest inventory of plutonium in double-shell tanks is in tank 241-102-SY.

To test the conservatism of this sample result, the tank 241-SY-102 concentration, 1.57×10^9 Bq/L (0.6968 g/L) was multiplied by the amount of sludge (268,764 L [71,000 gal]) reported to be in tank 241-SY-102 (November 1994). The result was 187.3 kg of plutonium. This is far more plutonium than could have gone to a single tank.

Sample data was available for 50% (12/24) of the nonaging waste double-shell tanks. The value used was an order of magnitude greater than the highest reported value from any other nonaging waste double-shell tank.

^{239/240}Pu in Aging Waste Facility Liquid - The data point used was 1.2×10^6 Bq/L, from a 1989 sample taken from tank 241-AZ-102.

Approximately 981 kg of plutonium were transferred to the tank farms, based on the recovery efficiency of the separations processes. Of this total, 83 kg were sent to double-shell tanks and the aging waste facility tanks. The review team was unable to identify a rationale for which of the four aging waste facility tanks would be expected to have the highest concentrations of plutonium on any basis other than sample results. The highest concentration based on sample results is from tank 241-AZ-102. The aging waste facility tanks contain the most recent waste from PUREX. Because it is PUREX waste, the most efficient process for separating plutonium, it is not unreasonable that the concentration of plutonium in the liquid is the lowest found in the three categories of tanks.

The four aging waste facility tanks all have large volumes of liquid that are free to mix and are arguably homogenous. The sample results from these tanks are expected to be representative of the bulk liquid in the tanks. Sample data was available for 100% (four of four) of the aging waste facility tanks. The value used was 2.7 times greater than the highest reported value from any other aging waste facility tank.

^{239/240}Pu in Aging Waste Facility Solids - The data point used was 4.4×10^8 Bq/L, from a 1989 sample taken from tank 241-AZ-101.

Approximately 981 kg of plutonium were transferred to the tank farms, based on the recovery efficiency of the separations processes. Of this total 83 kg were sent to double-shell tanks. The waste from PUREX, which contained the 83 kg, was sent to approximately 12 tanks in the AP, AW, AY, and AZ tank farms. Tank 241-AZ-101 is essentially filled with a single waste type. The airlift circulators were operating when tank 241-AZ-101 was filled, and continued to operate for several years. The solids are expected to be mixed and reasonably homogenous.

To test the conservatism of the sample concentration from tank 241-AZ-101, the sample was multiplied by the volume of sludge in the tank (0.1916 g/L x 132,489 L). The result was 25.4 kg of plutonium. It is not possible to say that a higher sample concentration will not be found. However, it is reasonable to say that a concentration that predicts 25.4 kg, out of a total of 83 kg, in one tank of a group of 12, is a very conservative number.

Sample data was available for 75% (3/4) of the aging waste facility tanks. The concentration used was 1.6 times greater than the next highest value.

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APPENDIX D
STATISTICAL DESCRIPTION OF RADIONUCLIDES IN TANK SAMPLES

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APPENDIX D

STATISTICAL DESCRIPTION OF RADIONUCLIDES IN TANK SAMPLES

D1.0 OVERVIEW

This appendix provides a statistical description of radionuclides in the population of sampled values used for calculating unit liter doses (ULDs) for WHC-SD-WM-SAR-067, *Tank Waste Remediation System (TWRS) Final Safety Analysis Report (FSAR)*, accident analyses. Analyses were performed for ^{241}Am , ^{137}Cs , ^{90}Sr , and $^{239/240}\text{Pu}$. The sample data used in this analysis was taken from WHC-SD-WM-ER-400, *Tank Waste Source Term Inventory Validation*.

Various features of the statistical data for the ^{241}Am , ^{137}Cs , ^{90}Sr , and $^{239/240}\text{Pu}$ analyses are given in tabular form in Tables D-1 through D-8 and in graphical form by way of histograms in Figures D-1 through D-24. The vertical line on each histogram indicates the mean of the data.

A quick overview of the tabular summaries reveals that the estimated 99th percentile of the data is of the same order of magnitude as the selected concentration used in the ULD. The selected concentration is typically up to four times larger than the 95th percentile.

A linear interpolation method, based on the largest observations, was used to generate the percentile estimates. An alternative method that involves fitting a gamma distribution to the entire dataset was also investigated. Comparisons of the percentile methods are provided in Tables D-5 through D-8.

D2.0 METHODS

The statistical parameters presented in the attached tables and figures were generated in a series of steps. First, for a given combination of radionuclide, storage facility, and state (solid or liquid), all of the Historical Tank Content Estimate (HTCE) (WHC-SD-WM-DTR-026, *Laboratory Characterization of Samples Taken in December 1991 (Window E) from Hanford Waste Tank 241-SY-101*; WHC-SD-WM-ER-349, *Historical Tank Content Estimate for the Northeast Quadrant of the Hanford 200 East Area*; WHC-SD-WM-ER-350, *Historical Tank Content Estimate for the Southeast Quadrant of the Hanford 200 East Area*; WHC-SD-WM-ER-351, *Historical Tank Content Estimate for the Northwest Quadrant of the Hanford 200 West Area*; WHC-SD-WM-ER-352, *Historical Tank Content Estimate for the Southwest Quadrant of the Hanford 200 West Area*) and track radioactive components (TRAC) (WHC-SD-WM-TI-057, *TRAC: A Preliminary Estimation of the Waste Inventories in Hanford Tanks Through 1980*) data was removed from the database. All observations from the tank characterization database (TCD) marked as "DUPLICATE" or "DUPLICATE RESULTS" were then removed. Using the average of the primary and duplicate entries would have been

preferable; however, due to time constraints, only the primary entries were used. Finally, if there were multiple entries with the same sample identification field, only the largest entry was retained. The values selected for ULDs were included; all of the larger values that had been examined and judged erroneous or inappropriate were removed as described in Section 3.3 of base document.

The mean is the arithmetic average of the data. The median is defined as the point with the property that half the data is smaller than it, and half is larger. The percentiles were linearly interpolated from the data subject to the restriction that they could not be larger than the largest observed value, using the S-plus quantile function. The last column of each table gives the selected concentration.

To provide perspective on the data, an alternate approach to estimating the percentiles was used for several datasets. The histograms for many of the datasets exhibit a shape similar to a gamma distribution. A gamma distribution was fit to a dataset using the method of moments (the shape parameter is estimated by the mean squared over the variance, and the scale parameter is estimated by the mean over the variance). Matrixes showing the comparison of the quantile estimates are provided in Tables D-5 through D-8. For the most part, the estimates are in reasonably close agreement. On the one hand, the estimates based on the gamma distribution are not constrained to be smaller than the largest observed value, like the empirically based, linearly interpolated estimates are. However, the estimates based on the gamma distribution are really only based upon the mean and the standard deviation of the data, and the bulk of the data in the smaller range has a huge impact on these two quantities. Thus, for the gamma estimates, it is the smaller data entries and the assumption of the shape of the gamma distribution that are determining the estimate. For these reasons, the empirical, linearly interpolated estimates are deemed more reliable.

It should be noted that the observations in each dataset have not been screened for errors and relevance, except for the selected source term value. This is because the screening process (obtaining original lab reports, checking for discrepancies, etc.) is very time consuming. Another difficulty is the fact that some tanks are sampled more than others; thus the population being considered is weighted toward those tanks that are sampled more often. This should be taken into account if the results given here are to be used in estimating the distribution of values expected in the general population of tanks. However, if tanks with higher concentrations of the radionuclides are sampled more often, use of this data to make conclusions about the general population of tanks will produce conservative results.

D3.0 REFERENCES

WHC-SD-WM-DTR-026, 1992, *Laboratory Characterization of Samples Taken in December 1991 (Window E) from Hanford Waste Tank 241-SY-101*, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

- WHC-SD-WM-ER-349, 1994, *Historical Tank Content Estimate for the Northeast Quadrant of the Hanford 200 East Area*, Rev. 0, Westinghouse Hanford Company, Richland, Washington.
- WHC-SD-WM-ER-350, 1995, *Historical Tank Content Estimate for the Southeast Quadrant of the Hanford 200 East Area*, Westinghouse Hanford Company, Richland, Washington.
- WHC-SD-WM-ER-351, 1995, *Historical Tank Content Estimate for the Northwest Quadrant of the Hanford 200 West Area*, Westinghouse Hanford Company, Richland, Washington.
- WHC-SD-WM-ER-352, 1994, *Historical Tank Content Estimate for the Southwest Quadrant of the Hanford 200 West Area*, Rev. 0, Westinghouse Hanford Company, Richland, Washington.
- WHC-SD-WM-ER-400, 1995, *Tank Waste Source Term Inventory Validation*, Rev. 0, Volumes I and II, Westinghouse Hanford Company, Richland, Washington.
- WHC-SD-WM-SAR-067, 1996, *Tank Waste Remediation System (TWRS) Final Safety Analysis Report (FSAR)*, DRAFT, Westinghouse Hanford Company, Richland, Washington.
- WHC-SD-WM-TI-057, 1984, *TRAC: A Preliminary Estimation of the Waste Inventories in Hanford Tanks Through 1980*, Rockwell Hanford Operations, Richland, Washington.

Table D-1. Summary statistics for ^{241}Am .

	Mean	Median	S.D.	95%	99%	Source
AWF solid	1.95 E+09	9.35 E+08	2.55 E+09	5.55 E+09	9.81 E+09	1.13 E+10
AWF liquid	2.00 E+05	5.63 E+04	3.06 E+05	6.10 E+05	1.00 E+06	1.10 E+06
SST solid	1.51 E+07	4.30 E+06	2.67 E+07	6.17 E+07	1.40 E+08	2.04 E+08
SST liquid	1.63 E+06	2.45 E+04	5.17 E+06	9.60 E+06	3.16 E+07	3.66 E+07
DST solid	1.27 E+08	5.23 E+06	3.97 E+08	8.93 E+08	1.69 E+09	2.97 E+09
DST liquid	2.00 E+06	2.68 E+04	6.13 E+06	1.05 E+07	2.89 E+07	3.19 E+07

All units are Bq/L, decayed. The "Source" column gives the selected source term value.

AWF = aging waste facility.

DST = double-shell tank.

S.D. = standard deviation.

SST = single-shell tank.

Table D-2. Summary statistics for ^{137}Cs .

	Mean	Median	S.D.	95%	99%	Source
AWF solid	2.55 E+10	1.04 E+10	3.03 E+10	1.00 E+11	1.02 E+11	1.02 E+11
AWF liquid	2.15 E+10	1.11 E+10	2.30 E+10	7.07 E+10	7.30 E+10	9.07 E+10
SST solid	7.13 E+09	1.91 E+09	1.24 E+10	3.00 E+10	6.15 E+10	7.47 E+10
SST liquid	6.20 E+09	4.01 E+09	6.12 E+09	1.76 E+10	2.11 E+10	2.25 E+10
DST solid	1.49 E+10	1.72 E+10	1.11 E+10	2.78 E+10	5.01 E+10	6.00 E+10
DST liquid	1.16 E+10	1.11 E+10	9.70 E+09	2.60 E+10	4.66 E+10	6.00 E+10

All units are Bq/L, decayed. The "Source" column gives the selected source term value.

AWF = aging waste facility.

DST = double-shell tank.

S.D. = standard deviation.

SST = single-shell tank.

Table D-3. Summary statistics for $^{239/240}\text{Pu}$.

	Mean	Median	S.D.	95%	99%	Source
AWF solid	1.14 E+08	6.36 E+07	1.17 E+08	2.71 E+08	4.06 E+08	4.42 E+08
AWF liquid	1.31 E+05	1.51 E+04	2.68 E+05	4.38 E+05	1.04 E+06	1.21 E+06
SST solid	2.96 E+07	5.24 E+06	6.38 E+07	1.60 E+08	3.23 E+08	4.45 E+08
SST liquid	1.78 E+06	1.57 E+05	4.92 E+06	6.62 E+06	3.20 E+07	3.62 E+07
DST solid	7.06 E+07	2.60 E+07	2.02 E+08	2.01 E+08	8.74 E+08	1.57 E+09
DST liquid	7.35 E+05	3.00 E+04	1.64 E+06	4.20 E+06	7.64 E+06	7.65 E+06

All units are Bq/L, decayed.
AWF = aging waste facility.

DST = double-shell tank.

S.D. = standard deviation.

SST = single-shell tank.

The "Source" column gives the selected source term value.

Table D-4. Summary statistics for ^{90}Sr .

	Mean	Median	S.D.	95%	99%	Source
AWF solid	4.94 E+11	1.49 E+11	6.81 E+11	1.51 E+12	2.40 E+12	2.99 E+12
AWF liquid	7.82 E+08	7.89 E+07	1.50 E+09	4.44 E+09	5.28 E+09	5.84 E+09
SST solid	8.59 E+10	4.39 E+09	2.27 E+11	5.77 E+11	1.10 E+12	1.70 E+12
SST liquid	1.15 E+09	2.05 E+08	2.03 E+09	5.43 E+09	1.04 E+10	1.11 E+10
DST solid	2.36 E+09	3.32 E+08	5.52 E+09	1.15 E+10	2.74 E+10	3.57 E+10
DST liquid	4.86 E+08	8.26 E+07	9.41 E+08	1.72 E+09	4.50 E+09	4.79 E+09

All units are Bq/L, decayed.

AWF = aging waste facility.

DST = double-shell tank.

S.D. = standard deviation.

SST = single-shell tank.

The "Source" column gives the selected source term value.

Table D-5. ^{241}Am - Comparison of Empirical, Linearly Interpolated Versus Gamma Distribution Percentile Estimates.

	Empirical 95%	Gamma 95%	Empirical 99%	Gamma 99%
AWF solid	5.55 E+09	7.08 E+09	9.81 E+09	1.19 E+10
AWF liquid	6.10 E+05	8.12 E+05	1.00 E+06	1.44 E+06
SST solid	6.17 E+07	6.76 E+07	1.40 E+08	1.28 E+08
SST liquid	9.60 E+06	9.46 E+06	3.16 E+07	2.50 E+07
DST solid	8.93 E+08	7.36 E+08	1.69 E+09	2.00 E+09
DST liquid	1.05 E+07	1.15 E+07	2.89 E+07	3.08 E+07

The empirical estimates are the ones given in Table D-1; the Gamma estimates are based on fitting a Gamma distribution to the data so that the mean and variance of the distribution match the observed data.

AWF = aging waste facility.

DST = double-shell tank.

SST = single-shell tank.

Table D-6. ^{137}Cs - Comparison of Empirical, Linearly Interpolated Versus Gamma Distribution Percentile Estimates.

	Empirical 95%	Gamma 95%	Empirical 99%	Gamma 99%
AWF solid	1.00 E+11	8.64 E+10	1.02 E+11	1.40 E+11
AWF liquid	7.07 E+10	6.76 E+10	7.30 E+10	1.06 E+11
SST solid	3.00 E+10	3.16 E+10	6.15 E+10	5.95 E+10
SST liquid	1.76 E+10	1.84 E+10	2.11 E+10	2.82 E+10
DST solid	2.78 E+10	3.65 E+10	5.01 E+10	5.18 E+10
DST liquid	2.60 E+10	3.03 E+10	4.66 E+10	4.48 E+10

The empirical estimates are the ones given in Table D-2; the Gamma estimates are based on fitting a Gamma distribution to the data so that the mean and variance of the distribution match the observed data.

AWF = aging waste facility.

DST = double-shell tank.

SST = single-shell tank.

Table D-7. ^{239/240}Pu - Comparison of Empirical, Linearly Interpolated Versus Gamma Distribution Percentile Estimates.

	Empirical 95%	Gamma 95%	Empirical 99%	Gamma 99%
AWF solid	2.71 E+08	3.47 E+08	4.06 E+08	5.38 E+08
AWF liquid	4.38 E+05	6.42 E+05	1.04 E+06	1.31 E+06
SST solid	1.60 E+08	1.50 E+08	3.23 E+08	3.13 E+08
SST liquid	6.62 E+06	1.00 E+07	3.20 E+07	2.46 E+07
DST solid	2.01 E+08	4.02 E+08	8.74 E+08	1.01 E+09
DST liquid	4.20 E+06	3.78 E+06	7.64 E+06	8.05 E+06

The empirical estimates are the ones given in Table D-3; the Gamma estimates are based on fitting a Gamma distribution to the data so that the mean and variance of the distribution match the observed data.

AWF = aging waste facility.

DST = double-shell tank.

SST = single-shell tank.

Table D-8. ⁹⁰Sr - Comparison of Empirical, Linearly Interpolated Versus Gamma Distribution Percentile Estimates.

	Empirical 95%	Gamma 95%	Empirical 99%	Gamma 99%
AWF solid	1.51 E+12	1.87 E+12	2.40 E+12	3.19 E+12
AWF liquid	4.44 E+09	3.68 E+09	5.28 E+09	7.24 E+09
SST solid	5.77 E+11	4.77 E+11	1.10 E+12	1.13 E+12
SST liquid	5.43 E+09	5.16 E+09	1.04 E+10	9.78 E+09
DST solid	1.15 E+10	1.24 E+10	2.74 E+10	2.73 E+10
DST liquid	1.72 E+09	2.31 E+09	4.50 E+09	4.57 E+09

The empirical estimates are the ones given in Table D-4; the Gamma estimates are based on fitting a Gamma distribution to the data so that the mean and variance of the distribution match the observed data.

AWF = aging waste facility.

DST = double-shell tank.

SST = single-shell tank.

Figure D-1. Americium-241 Aging Waste Facility Solid.

7G06030032.17
SRP-RAD

Figure D-2. Americium-241 Aging Waste Facility Liquid.

Figure D-3. Americium-241 Single-Shell Tank Solid.

Figure D-4. Americium-241 Single-Shell Tank Liquid.

Figure D-5. Americium-241 Double-Shell Tank Solid.

Figure D-6. Americium-241 Double-Shell Tank Liquid.

7G98030032.22
SRP-RAD

Figure D-7. Cesium-137 Aging Waste Facility Solid.

Figure D-8. Cesium-137 Aging Waste Facility Liquid.

Figure D-9. Cesium-137 Single-Shell Tank Solid.

7G06030032.31
SRP-RAD

Figure D-10. Cesium-137 Single-Shell Tank Liquid.

Figure D-11. Cesium-137 Double-Shell Tank Solid.

Figure D-12. Cesium-137 Double-Shell Tank Liquid.

7G98030032.34
SRP-RAD

Figure D-13. Plutonium-239/240 Aging Waste Facility Solid.

7G06030032.23
SRP-RAD

Figure D-14. Plutonium-239/240 Aging Waste Facility Liquid.

Figure D-15. Plutonium-239/240 Single-Shell Tank Solid.

Figure D-16. Plutonium-239/240 Single-Shell Tank Liquid.

Figure D-17. Plutonium-239/240 Double-Shell Tank Solid.

Figure D-18. Plutonium-239/240 Double-Shell Tank Liquid.

Figure D-20. Strontium-90 Aging Waste Facility Liquid.

Figure D-21. Strontium-90 Single-Shell Tank Solid.

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Figure D-22. Strontium-90 Single-Shell Tank Liquid.

Figure D-24. Strontium-90 Double-Shell Tank Liquid.

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