HNF-8741
Revision 0

T Plant Complex Radiological Characterization Program Plan

Prepared for the U.S. Department of Energy
Assistant Secretary for Environmental Management
Project Hanford Management Contractor for the
U.S. Department of Energy under Contract DE-AC06-96RL13200

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GLOSSARY

AEA	alpha energy analysis
CFR	Code of Federal Regulations
Ci	curie
GEA	gamma energy analysis
HNF	Hanford Site (document identifier1
ICP/MS	inductively coupled plasma-mass spectrometry
LSC	liquid scintillation counting
MCNP	Monte-Carlo N-Particle (software)
PUREX	plutonium-uranium extraction
QA	quality assurance
QC	quality control
REDOX	reduction oxidation
TRU	transuranic (waste)
TWINS	Tank Waste Inventory System (database)
WHC	Westinghouse Hanford Company
WMPQAPP	Waste Management Project Quality Assurance Program Plan

METRIC CONVERSION CHART

Into metric units			Out of metric units		
If you know	Multiply by	To get	If you know	Multiply by	To get
Length			**Length**		
inches	25.40	millimeters	millimeters	0.03937	inches
inches	2.54	centimeters	centimeters	0.393701	inches
feet	0.3048	meters	meters	3.28084	feet
yards	0.9144	meters	meters	1.0936	yards
miles (statute)	1.60934	kilometers	kilometers	0.62137	miles (statute)
Area			**Area**		
square inches	6.4516	square centimeters	square centimeters	0.155	square inches
square feet	0.09290304	square meters	square meters	10.7639	square feet
square yards	0.8361274	square meters	square meters	1.19599	square yards
square miles	2.59	square kilometers	square kilometers	0.386102	square miles
acres	0.404687	hectares	hectares	2.47104	acres
Mass (weight)			**Mass (weight)**		
ounces (avoir)	28.34952	grams	grams	0.035274	ounces (avoir)
pounds	0.45359237	kilograms	kilograms	2.204623	pounds (avoir)
tons (short)	0.9071847	tons (metric)	tons (metric)	1.1023	tons (short)
Volume			**Volume**		
ounces (U.S., liquid)	29.57353	milliliters	milliliters	0.033814	ounces (U.S., liquid)
quarts (U.S., liquid)	0.9463529	liters	liters	1.0567	quarts (U.S., liquid)
gallons (U.S., liquid)	3.7854	liters	liters	0.26417	gallons (U.S., liquid)
cubic feet	0.02831685	cubic meters	cubic meters	35.3147	cubic feet
cubic yards	0.7645549	cubic meters	cubic meters	1.308	cubic yards
Temperature			**Temperature**		
Fahrenheit	subtract 32 then multiply by 5/9ths	Celsius	Celsius	multiply by 9/5ths, then add 32	Fahrenheit
Energy			**Energy**		
kilowatt hour	3,412	British thermal unit	British thermal unit	0.000293	kilowatt hour
kilowatt	0.94782	British thermal unit per second	British thermal unit per second	1.055	kilowatt
Force/Pressure			**Force/Pressure**		
pounds (force) per square inch	6.894757	kilopascals	kilopascals	0.14504	pounds per square inch

06/2001

Source: ***Engineering*** *Unit Conversions*, M. R. Lindeburg, PE.. Third Ed., 1990. Professional Publications. Inc., Belmont, California.

T PLANT COMPLEX RADIOLOGICAL CHARACTERIZATION PROGRAM PLAN

1.0 INTRODUCTION

This plan describes the methods used to identify and quantify the radionuclide content of radioactive waste managed at T Plant Complex. A technical basis is provided for the selection of the major radionuclides in T Plant Complex waste streams. From this basis, specific characterization strategies and methods are identified.

1.1 BACKGROUND

T Plant Complex, located in the 200 West Area of the Hanford Site, was constructed in 1943 as the first separations facility to extract plutonium for weapons from uranium fuel irradiated in Hanford Site nuclear reactors. Plutonium was extracted in the 221-T Canyon Building by dissolving the irradiated uranium fuel and using a series ofprecipitation reactions known as the bismuth phosphate process. Uranium and fission products were discharged to onsite high-level waste tanks, while solutions of the extracted plutonium were transferred to the 224-T Facility to be concentrated. The 221-T Facility carried out this plutonium separation mission from 1944 until March 1956, at which time the Reduction Oxidation (REDOX) Plant and Plutonium-Uranium Extraction (PUREX) Facility provided adequate capacity and more efficient methods to separate plutonium

Shortly after shutdown, the 221-T Canyon Building process cells were washed down and the process equipment was decontaminated (WHC-MR-0452). Equipment was removed from a number of the process cells to prepare T Plant Complex for a new mission to decontaminate equipment from other Hanford Site facilities. A wide variety of decontamination methods were used, including high-pressure water, sand blasting, vibratory finishing, and chemical solutions. Decontamination was performed in process cells and on the canyon deck. In 1959, the 2706-T Building was constructed for decontamination of equipment that was too large to be handled in the 221-T Canyon Building and for smaller equipment that was not highly contaminated. In the early 1990's, T Plant Complex began accepting radioactive waste from other facilities for characterization, correction of packaging and segregation problems, and waste treatment. Currently. the primary mission of T Plant Complex is management of radioactive waste, with a major emphasis on preparing to store remote-handled transuranic (TRU) sludge from the 105-K Basins.

1.2 REQUIREMENTS

Radiological characterization requirements for waste are derived from two primary sources: DOE Order 435.1, "Radioactive Waste Management". and the Hazardous Materials Transportation Regulations of **49** Code of Federal Regulations (CFR) and corresponding Hanford Site requirements.

DOE Order 435.1 establishes general requirements that define the type and extent ofradiological characterization required. In particular, two provisions *of* the Order define characterization requirements. First. the Order requires that facilities that manage radioactive waste have established acceptance criteria. These acceptance criteria must include limits on allowable radionuclide concentrations based on facility safety analyses and performance assessments. Second, the Order specifies that the data quality objectives process or a similar process be used to ensure that the characterization data are of adequate quality to make the specific required decisions (e.g., does the waste meet an acceptance criteria limit?). For the Hanford Site, the *Hanford Sire Solid Waste Acceptance Criteria* (HNF-EP-0063, current revision)

identifies the applicable facility limits. Additionally. HNF-EP-0063, Section 2.4, establishes acceptable approaches to radiological characterization.

The 49 CFR regulations establish classification methods and limits for transportation of radioactive material. On tlie Hanford Site. waste transfers must either meet 49 CFR requirements or must be evaluated under a packaging safety analysis that demonstrates **a** level of safety comparable to the requirements of 49 CFR. **A** facility's radiological characterization methods must provide sufficient knowledge of the radionuclides in the waste to properly classify the waste for transfer.

The characterization strategies and methods described in this plan have been written to address both the facility limits of HNF-EP-0063 and the applicable transportation requirements.

1.3 GENERAL DESCRIPTION OF RADIOACTIVE MATERIAL IN T PLANT COMPLEX WASTE STREAMS

The T Plant Complex consists of a number of facilities that are in active use. including both radiological and nonradiological facilities. Radioactive waste is generated primarily from the 221-T Canyon Building. and associated structures (221-T railroad tunnel, 221-TA annex, 291-T filtration system) and the 2706-T Building, where active waste management and decontamination activities occur. Radioactive material **is** stored on various pads and in the 214-T Storage Building, but little radioactive waste **is** generated from these facilities, which are operated as radiologically clean facilities.

Radiological characterization of T Plant Complex waste streams is complicated by the fact that there are a variety of sources of radioactive material. In general terms, these sources can be classified as 'legacy' sources (i.e., radioactive material from past activities at T Plant Complex, including the original bismuth phosphate process and the various decontamination and waste management activities) and 'project' sources (i.e., radioactive material in waste from other facilities that is being processed or treated at T Plant Complex). The characterization methodology for these two sources of radioactive material differs significantly, as described later in this document. The following is a brief description of the types of radioactive material from legacy and current project sources.

1.3.1 Legacy Contamination in the 221-T Canyon Building

The 221-T Canyon Building is contaminated with residues remaining from the bismuth phosphate process and from subsequent decontamination activities perfomied in the canyon. From the beginning of T Plant Complex operations in 1944 until 1956, the process cells were used lo extract plutonium from spent nuclear fuel from Hanford Site single pass reactors. An overall source term for the fuel processed at T Plant Complex has been published (HNF-SD-WM-TI-794). This source term provides a reasonably reliable list of expected radionuclides and the relative distribution in unprocessed spent fuel, decay corrected to 1994. The average distribution ofradionuclides from spent fuel processing at T Plant Complex is provided in Table 1, column 2.

This spent fuel was dissolved and the plutonium separated by tlie bismuth phosphate process, described in detail in the Hanford Technical Manual. Section C (HW-10475-C) and many subsequent documents. The fuel cladding was removed and the uranium fuel dissolved toward the head end of the 221-T Canyon Building (cells 3 through 6). In cells 7 through 11, an extraction was performed to separate the plutonium from the bulk of the uranium and fission products in the fuel. Uranium and fission product waste was treated and discharged to onsite **Tank** Farms. The crude plutonium product was purified further in a series of decontamination steps in cells 12 through 19. The waste from these decontamination steps was treated and discharged to Tank Farms. Generalized data are available in HW-10475-C regarding the

separation of fission products and plutonium at each process step. The distribution of isotopes in the early process cells would he very similar to that of Table 1, column 2. Moving down the canyon, the same radionuclides would be present, hut the fission products became increasingly dilute as the plutonium product **was** extracted.

Subsequent to 1956, T Plant Complex was used for decontamination of equipment from other Hanford Site facilities, primarily the Tank Farms, PUREX Facility, and B Plant Complex. The radionuclides of concern for these facilities would he expected to be very similar to those of T Plant Complex, **as all** processing activities were related to the spent fuel from onsite reactors. The relative distribution of these radionuclides, however, varies significantly among these facilities. HNF-SD-WM-TI-794 provides an overall source term for fuel processed through the PUREX Facility, REDOX Plant, and B Plant Complex. Extensive data are available on the source term for the Tank Fanns in the Tank Waste Inventory System (TWINS) database and other sources. The later Cs-137 and Sr-90 extraction mission at B Plant Complex and the uranium recovery process at U Plant would have similar isotopes of concern **as** those identified in HNF-SD-WM-TI-794, hut would he expected to vary considerably in their expected distribution. For illustrative purposes, columns 3 and 4 of Table 1 provide average radionuclide distributions for all fuel processed through separations plants and for Hanford Site tank waste.

From the sources discussed previously. it is clear that Cs-137 and Sr-90 represent the majority of the radiological activity in typical waste streams. In spent fuel processed through T Plant Complex, for example, Cs-137 and Sr-90 make up roughly 96% of the radiological activity, with plutonium and Am-241 making up an additional 3% ofthe total activity. Based on these data, Cs-137 represents well over 90% of the gamma emissions from expected waste streams, and thus serves as a good indicator isotope for quantification of total activity.

1.3.2 Legacy Contamination in the **291-T** Exhaust System

The 29 I-T exhaust system provides ventilation of the 221-T Canyon Building. The source term is based on dispersible radioactive material from activities and legacy Contamination in the 221-T Canyon Building. Current project activities are operated in a relatively clean manner. so new sources are believed to provide an insignificant portion ofthe contamination in the 291-T exhaust system. The source term for waste derived from the 291-T system is expected to he similar to that ofthe 221-T Canyon Building, as discussed in the previous section.

1.3.3 Legacy Contamination in the **2706-T** Building

The 2706-T Building has been used for decontamination of large equipment and processing of waste that, is not highly radioactive. Tile majority of the 7706-T Building is essentially free of legacy contamination. The 2706-T Building was decontaminated and upgraded during the late 1990s. During these upgrades, potential sources of legacy contamination were cleaned and sealed. The original ventilation system (ACT-I) retains some contamination from previous decontamination and waste processing activities. This legacy contamination comes from sources similar to the decontamination activities performed in the 22 I-T Canyon Building, although in much lower concentrations.

1.3.4 **Radioactive** hlaterials from Current Activities (Project Waste)

The T Plant Complex organization routinely performs projects, including waste processing, treatment, additional characterization. and decontamination. At this time, these activities generally are performed in

the 2706-T Building. However, the 221-T canyon and tunnel also are used as needed, particularly for more radioactive waste.

Waste that is accepted at T Plant Complex for a project either has full radiological characterization data provided by the generator, or has partial characterization data that is to be completed through sampling and analysis perfonned as part of the project. As a result, the radiological source term associated with project waste is established from generator-provided data and. when specified, additional data obtained during characterization at T Plant Complex.

1.4 EVALUATION OF MAJOR RADIONUCLIDES T PLANT COMPLEX WASTE STREAMS

Section 1.3 provided *a* broad overview of the types of radioactive material present at T Plant Complex. This section provides an evaluation of these radionuclides against applicable disposal and transportation limits to identify the radionuclides of primary interest in T Plant Complex waste streams. Note that this evaluation applies only to legacy contamination at T Plant Comples. Project waste accepted from other facilities might have other radionuclides of interest. which are identified in the waste acceptance documentation.

An evaluation of the legacy source terms was perfonned to determine which isotopes could he the radionuclides of primary interest in T Plant Complex waste streams. The average isotopic distribution from all Hanford Site fuel (Table 1, column 3) was selected for evaluation against key limits of HNF-EP-0063, Revision 6, and 49 CFR. A reasonable bounding case for contact-handled T Plant Complex waste was evaluated: a 5 foot by 5 foot by 9 foot metal box containing 1 Ci of Cs-137. This case was evaluated against the mobile radionuclide reporting limits and Category I and Category 3 limits of HNF-EP-0063, Revision 6, and against the A2 values of 49 CFR. The results of this case are provided in Table 2 3s percentages of the applicable limits.

As a general rule, radionuclides of interest for disposal are those that exceed the mobile radionuclide reporting limit or exceed 1% of the Category 1 limit. For transportation, 49 CFR requires reporting of the set of isotopes that make up 95% of the cumulative A2 fraction in the waste. Table *2* shows that only a limited set ofradionuclides are of significance for transportation and disposal under this case'. Specifically, Sr-90, Sn-126. Cs-137, the various isotopes of uranium and plutonium, and An-241 could be significant in this bounding case. C-14, Se-79, Tc-99, I-129, and Np-237 are below the mobile radionuclide reporting limit for the bounding case, and could he significant only in waste streams with an isotopic distribution that significantly is altered from the total Hanford Site fuel distribution. Other isotopes listed in Table 2 are unlikely to be significant with respect to any of the transportation or disposal limits in legacy-contaminated waste streams'.

It is understood that the actual distribution of radionuclides in T Plant Coinplex waste streams could be significantly altered from that presented in Table 1, column 3. Changes in this distribution are because of two primary factors: (1) intentional separation of certain isotopes at T Plant Complex and other

[1] Significant isotopes for HNF-EP-0063, Revision 6. would include those that exceed the mobile radionuclide reporting limit or exceed 1% of the Category 1 limit. Significant isotopes for transportation would be those that contribute 95% of the cumulative A2s; for this cursory evaluation, any individual isotope that contributes 1% of an A2 could he considered to be significant.

[2] However, other isotopes. might be reportable in some waste packages under the reporting criteria of HNF-EP-0063. Revision 6, Section 2.4. For example, sill-151 might contribute more than 1% of the total activity in a waste package (Table 1) and be reportable even though Sm-151 does not approach any specific limits.

separations plants, and (2) differences in solubility of the chemical species in which the isotopes commonly occur. The first factor is the most significant. T Plant Complex and the other separations buildings separated and purified plutonium from uranium and other fission products. As a result, materials contaminated with plutonium product solutions would have much more plutonium relative to uranium and fission products, while waste solutions would have much lower concentrations of plutonium, as indicated in Table 1, column 4. Similarly, the PUREX Facility separated uranium for recycling, U Plant recovered uranium from tank waste, and B Plant Complex recovered Cs-137 and Sr-90 from tank waste. Waste streams from these sources would have a similar set of isotopes as provided in Table 1, column 3, but some streams significantly would have altered distributions of those isotopes.

Chemical solubility also can alter the distribntion of waste streams. In particular, most transition metals, actinides, and lanthanides largely are insoluble in neutral to basic solutions. Cesium, an alkali metal, is very soluble in most chemical forms, regardless of pH. Nonmetal radionuclides (e.g., H-3, C-14, I-129) typically occur in chemical forms that relatively are soluble. **As a** result, waste streams that derive the radioactive material from settled sludge in tanks or have been washed extensively will have a higher proportion of insoluble metal radionuclides (e.g., Sr-90, uranium, plutonium) relative to Cs-137 and other more soluble isotopes. This effect can be seen in sample data from T Plant Complex tanks and Tank Farms, where sludge and supernatant samples from the same tank significantly have different distribution of soluble and insoluble species.

To summarize, column 3 of Table 1 represents **a** global average distribution of radionuclides managed in T Plant Complex and the other Hanford Site separations plants, and serves as the starting point for identifying isotopes of concern Intentional separation of uranium, plutonium, Cs-137, and 3-90 can cause large changes in the relative distribution; as a result, characterization methods must quantify the uranium, plutonium, Cs-137, and Sr-90 concentrations. The relative distribution of other fission products also will vary, though to a much lesser extent because of solubility factors rather than intentional separation. As a result, the remaining fission products can be characterized adequately using scaling factors from HNF-SD-WM-TI-794 or other applicable process knowledge documents. Those fission products that are well below the applicable limits of Table 2 would not be major radionuclides in T Plant Complex legacy contamination. As **a** result, the radionuclides of concern in T Plant Complex waste streams (other than those derived from project waste) are shown in Table 3.

2.0 CHARACTERIZATION STRATEGIES FOR MAJOR WASTE STREAMS

This section describes the characterization strategies used for major T Plant Complex waste streams. This section **is** not intended to provide detailed methods for characterization; instead, the discussion in this section identifies, in general terms. the most appropriate approaches that can be used to characterize waste streams. Additional detail on specific characterization methods is provided in Section 3.0.

A brief clarification is needed on the use of the term 'waste stream' in this document. The common meaning of waste stream is a waste or group of wastes from a process or a facility with similar physical, chemical, or radiological properties (HNF-EP-0063). Based on this definition, T Plant Complex generates a large number of waste streams. For the purposes of this document, however, the discussion is limited to the 'radiological' properties of the waste. **As** a result, different waste streams with similar radiological properties would be characterized using the same methodologies. As an example, F001 through F005 mixed waste debris from the 221-T Canyon Building deck is characterized in the same manner as low-level debris from the same source.

2.1 WASTE STREAMS FROM WASTE PROCESSING AND DECONTAMINATION PROJECTS

As described in Section 1.3.4, each container of waste accepted at T Plant Coinplex either has full radiological characterization, or has partial characterization that is completed during the project. The radiological characterization of waste generated from these projects is based on the inventory of the containers processed as part of the project.

For the purposes of radiological characterization, there are two types of waste streams from projects

- Primary project waste. This consists of the actual waste that was repackaged, transferred, treated, or otherwise processed. This type of waste is assumed to retain essentially all of the radionuclide inventory of the original container(s) from which the waste is derived. The only characterization required is to apportion the relative fraction of the original activity to the new container(s).

- Secondary waste. Secondary waste includes personnel protective equipment, tools. contarnination control supplies, and related materials that become incidentally contaminated with radioactive material from the processed waste. Only a small fraction of the radioactive material processed in a project contaminates the secondary waste.

2.1.1 Characterization of Primary Project Waste

Primary project waste is characterized by simply apportioning tlie initial radionuclide inventory to the final waste container(s) for the project. The relative fraction of the initial waste mass in a given waste container is used to estimate the fraction of the initial activity in that waste container. In some cases, the majority of the activity might be concentrated in a specific portion of tlie original waste (e.g., sealed sources, inner containers in a lab pack). In such cases. the activity should be apportioned in accordance with the process knowledge provided by the generator.

2.1.2 Characterization of Secondary Waste from Projects

Secondary waste from projects contains only a small fraction of the activity of the primary waste that was processed. For secondary waste stream. the radionuclide concentration can be detennined as follows.

- Bounding estimate of fraction of project activity in waste. For some projects, it is possible to establish a reliable bounding estimate of the fraction of tlie initial radioisotope inventory in the secondary waste. These estimates can be based on contamination surveys or knowledge of the activity being performed. While not an ideal approach, cases where the initial activity of the project waste are sufficiently low could legitimately use an extremely conservative bounding estimate that 100% of the original activity was transferred to the secondary waste.

  Example: A project involves emptying glass jars of waste into larger containers. The amount of residue in the gloss jar is less than 1%, so a bounding estimate is made that 1% of the initial activity remains on the empty containers.

- Dose to activity modeling with project-specific scaling factors. If Cs-137 is responsible for at least 90% of the gamma activity of the isotope mixture, dose to activity modeling can be used to establish the total activity of Cs-137 in each waste container. The initial radionuclide inventory can be used to establish scaling factors relating Cs-137 to the other radionuclides in tlie waste. Dose to activity

modeling is performed in accordance with Section 3.2. Calculation of the isotopic inventory of a container is determined by

$$A_i = A_{Cs\text{-}137} \times SF_i$$

where A_i = activity of isotope i
$A_{Cs\text{-}137}$ = activity of Cs-137 detennined from dose to activity modeling
SF_i = scaling factor far isotope i.

The same approach could be used for other high-energy gamma emitting radionuclides (e.g., Co-60) as long as reliable scaling factors can be established and isotope-specific dose to activity modeling is performed.

2.2 WASTE STREAMS FROM THE 221-T CANYON BUILDING DECK AND TUNNEL

During the past 2 years, T Plant Complex has removed the majority of equipment and waste from the canyon deck and tunnel in preparation to receive sludge from the K Basins. As a result. in the future the majority of the waste from the canyon deck and tunnel will be step-off pad waste, supplies used for contamination control during operations, and varying forms of maintenance waste. These materials typically will have low levels of contamination that are difficult to measure using sampling and analysis.

Canyon deck and tunnel waste is characterized using dose to activity modeling in conjunction with scaling factors that relate the concentration ofthe radionuclides ofconcern to Cs-137. Dose to activity modeling is performed in accordance with Section 3.2. Calculation of the isotopic inventory of a container is detennined by

$$A_i = A_{Cs\text{-}137} \times SF_i$$

where A_i = activity of isotope i
$A_{Cs\text{-}137}$ = activity of Cs-137 detennined from dose to activity modeling
SF_i = scaling factor for isotope i.

T Plant Complex canyon and tunnel scaling factors are developed froin characterization sampling and analysis. along with process knowledge from HNF-SD-WM-TI-794, as described in Appendix A. Confirniation sampling and analysis from the canyon and tunnel should be performed every 3 years or when there is a significant change in canyon activities that could affect the distribution of radionuclides. The confirmation sampling data will be used to adjust the scaling factors as appropriate.

2.3 WASTE STREAMS FROM THE 221-T CANYON BUILDING PROCESS CELLS

Equipment and residues are being removed from some of the 221-T Canyon Building process cells in preparation for storage of K Basins sludge. The process cells represent a particularly difficult characterization challenge because of the high concentrations of radioactive material and significant heterogeneity of the isotopic distribution (particularly in terms of the proportion of TRU to fission products) as described in Section 1.3.1. Adequate management of waste relies on a two-step approach in which in-cell TRU characterization is used first to determine whether the waste is TRU or low-level waste (LLW). For the LLW portion of the waste, nondestructive assay (NDA) is performed after packaging to confirm the TRU determination and to establish an accurate isotopic inventory for transportation and

storage or disposal. Typically, TRU waste is transferred to other process cells for storage rather than packaging for storage at the Central Waste Complex.

2.3.1 In-Cell Transuranic Waste Characterization

In-cell TRU characterization involves screening analysis to differentiate TRU waste from LLW. Two primary methods can be used for this TRU characterization.

- Neutron flux measurements. The TRU concentration of waste items can be estimated by measuring the neutron flux at various locations in tlie process cell. This neutron flux data can serve as inputs to dose to activity modeling software to infer a TRU concentration. Neutron flux can be measured in a number of ways, including use of portable neutron counting panels and activation of copper coupons placed in the process cell followed by counting gamma emissions with sodium iodide detectors.

  The neutron flux measurements are most effective for characterization of large pieces of equipment of known geometry, but can be used for other materials as well.

- Sampling and radiochemical analysis. Cell residues can he sampled and submitted for radiochemical analysis to determine the TRU concentration of the waste. The minimum radiochemical analysis for such purposes would be gross alpha, isotopic plutonium. and gamma energy analysis. Other analyses typically would be performed for later use in characterization of packaged LLW (should the waste classify as LLW), including uranium and Sr-90 analyses.

2.3.2 Characterization of Packaged Waste

The TRU screening characterization does not provide sufficient data for disposal and transportation classification. As a result, LLW that is packaged in the canyon is characterized further for disposal purposes. Two methods can be used for characterization of packaged waste.

- Dose to activity modeling and scaling factors. This method is identical to the method used for canyon deck waste characterization, except that cell-specific or item-specific scaling factors must be developed. These scaling factors would be based on sampling and analysis data from the cell waste in conjunction with applicable data from HNF-SD-WM-TI-794.

- NDA. Nondestructive gamma-ray and neutron assay can be used in conjunction with bounding scaling factors. Gamma-ray assay provides a reliable measurement of the Cs-137 activity, along with any other high-energy gamma emitting isotopes in tlie waste. Neutron assay provides a reliable measurement ofthe total TRU isotope activity, which can be used to estimate the Pu and Am-241 activity in the packaged waste. The quantity of Sr-90 and other isotopes of concern would be based on sampling and analysis data from the cell waste in conjunction with applicable data from HNF-SD-WM-TI-794.

2.4 WASTE STEAMS FROM THE 2706-T BUILDING

The majority of waste generated in the 2706-T Building is project waste, which is characterized as specified in Section 2.3.1. High-efficiency particulate air (HEPA) filters are characterized in accordance with Section 2.5. Any maintenance waste streams or other waste that is not associated with project waste would require sampling and analysis perfoniied in accordance with Chapter 3.0, Section 3.3.2.

2.5 FACILITY VENTILATION FILTERS (HEPA AND PRE-FILTERS)

The 221-T Canyon Building and the 2706-T Building each have HEPA filtration systems. Pre-filters and HEPA filters are replaced periodically as these become loaded or fail. The 221-T Canyon Building filtration system is located in the 291-T Facility and consists of two banks of HEPA filters with pre-filters. The 2706-T Building has two systems, each having HEPA filters with pre-filters.

Characterization of filters and pre-filters is perfonned by sampling and analysis of the filter media and/or NDA. Either of the following methods can be used.

- Filter media sampling and analysis. Because of the high degree of mixing of particulates in the ventilation, a single sample is considered representative of the entire bank. Sampling and analysis data are used to establish the isotopic inventory as described in Table 4. The analysis result, expressed in units of activity per unit mass, is multiplied by the mass of the filter media to obtain the total radionuclide inventory of the filter.

- NDA. NDA can be used to quantify the Cs-137 activity, as well as other high-energy gamma emitting radionuclides that might be present. The Cs-137 activity is used in conjunction with facility scaling factors to establish the radionuclide inventory of the waste. For the 291-T filtration systems, the scaling factors for the 221-T Canyon Building can be used (Appendix A). For the 2706-T Building filtration systems, sampling and analysis must be performed to establish scaling factors; at a minimum, total alpha, total beta, and gamma energy analysis would be required.

2.6 TANK WASTE

T Plant Complex does not discharge actively to its' tank systems. **As** a result, disposal of tank waste is not a routine operation. When tank waste is transferred to the Tank Fanns high-level waste tanks, extensive radiochemical analysis must be performed in accordance with the double-shell tank waste acceptance criteria. This analysis data can he used to develop scaling factors for any solid waste contaminated with tank waste (e.g., sampling equipment).

2.7 OTHER WASTE STREAMS AND CHARACTERIZATION STRATEGIES

The previous sections describe characterization of the majority of the possible waste streams at T Plant Complex. When waste streams are generated that do not fit in the description of one of the previous sections, a stream-specific or project-specific characterization plan must be developed. If any of these streams are generated in an ongoing basis, the stream characterization method will he added to this document during the next update.

Similarly, it might not he feasible or effective to characterize some specific waste items using the strategies provided previously. In such cases, an item-specific or project-specific characterization plan must he developed.

3.0 CHARACTERIZATION METHODOLOGIES

This section describes the specific methodologies used to characterize T Plant Complex waste streams.

3.1 DOSE TO ACTIVITY MODELING

With the exception of waste streams from certain waste processing projects, T Plant Complex waste streams contain Cs-137 as a large fraction of the radiological activity. The Cs-137 activity readily can be quantified from dose rate data using commercially available software such as Microshield 5.0© and Monte-Carlo N-Particle (MCNP) software. Waste streams that contain other high energy gamma emitting isotopes (e.g., Co-60) also can use dose to activity modeling, with specific model inputs **for** the indicator radionuclide. This section describes the methods for dose to activity modeling when specified in Section 2.0.

For packaged waste in standard containers, dose to activity calculations already have been developed using MCNP and published in WHC-SD-WM-RPT-267. Dose to activity curves have been developed for 55-gallon drums, 85-gallon drums, 4 foot by 4 foot by 8 foot wood boxes, 5 foot by 5 foot by 9 foot metal boxes, as well as certain other container sizes that commonly are not used by T Plant Complex. For these standard containers, the dose to activity curves from WHC-SD-WM-RPT-267 are used to compute the Cs-137 activity. The following notes and limitations apply to use of this document.

- The procedure of Appendix A, Section 3.0 of WHC-SD-WM-RPT-267 must be followed, except that T Plant Complex-specific spreadsheets can be used for calculational purposes.

- Six-point dose rate surveys should be used for all container sires. WHC-SD-WM-RPT-267 allows use of 14-point surveys, but this survey method is not standard and will not be used.

- The waste weight must not exceed the highest amount modeled (i.e., the highest weight plotted on the dose to activity curves).

- Container sizes can vary slightly from the nominal dimensions modeled. For example. the dose to activity curve for a 4 foot by 4 foot by 8 foot box could be used for a 3.5 foot by 4 foot by 7.5 foot box.

- **All** measurements (dose rate, weight) will be documented.

Modeling of nonstandard containers or specific pieces of equipment will be performed on a case-by-case basis. For performance of custom dose to activity modeling, the following notes and limitations apply.

- The software used for modeling must be controlled and validated in accordance with applicable software quality assurance requirements.

- The individual performing modeling must be experienced and knowledgeable in the use and liinitations of the software.

- All assumptions used and inputs to the model clearly must be documented. These include, but are not limited to, dose rate measurements, background dose rates, location of dose rate measurements. container/equipment geometry (including drawings when appropriate). weight measurements, assumptions regarding material density, and assumptions regarding distribution of radioactive material in the container or equipment.

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- **A** report must be created that identifies all assumptions and inputs used to the model, along with the computer model output, and any calculations perfonned using other software.

3.2 SAMPLING AND RADIOCHEMICAL ANALYSIS

Radiochemical analysis is perfonned either for direct characterization of a specific waste item or a hatch of waste, or to develop scaling factors for a waste stream

3.2.1 Sampling

The sampling strategy must provide samples that are representative of the material that is being characterized. Sampling strategies can vary widely, depending on the waste matrix and use of the data. Specific sampling methods will be described in a sampling and analysis plan or other sampling instructions.

In obtaining representative samples, the following general requirements must he met.

- Smear samples can he used for characterization of hard surfaces where the majority of the radioactive contamination **is** believed to he readily removable.

- For hard surfaces where a significant portion of the radioactive contamination readily is not removable (e.g., painted items, corroded surfaces), the sampling method must remove the contaminated surface layer. Examples include removing paint with a putty knife or paint stripper and scraping corroded surfaces with a knife.

- Bulk samples should he obtained for liquids and readily samplable solids.

- Samples obtained to develop scaling factors should he obtained from areas of highest contamination on the waste that is being characterized. Field survey instruments can he used to determine areas of highest contamination.

- Samples obtained to directly determine the concentration ofradionuclides (i.e., not for scaling factors) should use a random sampling strategy to determine the average radionuclide concentration. Such samples can be analyzed individually or can be coinposited to reduce the number of analyses.

3.2.2 Analysis Methods

Analysis methods must provide reliable quantification of the radionuclides of interest at the required detection levels. Table 4 provides the standard methods available at Hanford Site analytical laboratories. Other methods can be used as long as the methods provide data of adequate quality for the analysis objectives.

3.3 CALCULATION OF SCALING FACTORS

Scaling factors for Sr-90. U-233, U-234, U-235. U-236, U-238, Pu-238, Pu-239, Pu-240, Pu-241, and An-241, and any isotopes detected by GEA will be computed from analysis data. Scaling factors for other isotopes will be determined from process knowledge. The following sections describe the methods that will be used to compute these scaling factors.

3.3.1 Scaling Factors from Analysis Data

Scaling factors are computed from analysis data by dividing the reported activity of each measured isotope by the reported Cs-137 activity of the same sample. In calculating scaling factors from analysis results, the following notes and assumptions should be followed.

- If Cs-137 is not detected in a sample, data from that sample are rejected and not used in computation of scaling factors.

- If results for isotopes of concern other than Cs-137 are below the detection limit. the reported limit of detection is used to form the scaling factor.

- For total alpha and total uranium results, the scaling factor will be determined using the most appropriate isotopic distributions of HNF-SD-WM-TI-794, or other appropriate process knowledge documentation.

- When multiple samples have been obtained for a given item or container, the scaling factor ordinarily is determined using the mean of the results. Either a simple average or a weighted average of the distributions can be used.

3.3.2 Scaling Factors from **Process Knowledge**

Scaling factors for other radionuclides are calculated from the appropriate process knowledge documents. Table 5 summarizes the most current and reliable data sources available.

4.0 QUALITY ASSURANCE

Quality assurance (QA) requirements for radiological characterization will meet the Waste Management Project Quality Assurance Program Plan (WMP QAPP), HNF-SD-WM-QAPP-036, current revision. Beyond the general requirements of the WMP QAPP, the following quality requirements will be implemented for waste characterization.

- Procurement of analytical services. When procuring analytical services (e.g., radiochemical analysis, NDA), the statement of work must identify QA and quality control (QC) requirements. The rigor of QA/QC will be in accordance with the graded approach.

- Software QA and calculations. **All** software used for calculations related to radiological characterization will be tested and controlled in accordance with current onsite requirements

- Records. The following characterization documents are quality records and will be retained in accordance with HNF-PRO-222, Quality Assurance Records Standards: laboratory analysis reports; NDA reports; any data used in radiological calculations, including dose to activity modeling; and radionuclide inventory and classification calculations.

5.0 RECERTIFICATION

A variety of factors could require revision of the characterization methods and data used for waste at T Plant Complex. These include changes of mission, major maintenance activities, significant decontamination effons, and decay and ingrowth of radionuclides. To ensure the characterization methods remain appropriate, this characterization plan must be evaluated every 3 years, at a minimum, and updated as needed. In addition, whenever there is a major change in radiological conditions, the plan must be evaluated and updated as needed.

6.0 REFERENCES

HNF-EP-0063, *Hanford Site Solid Waste Acceptance Criteria*, Fluor Hanford, Richland. Washington. 2001 (revised periodically).

HNF-PRO-222, *Quality Assiirunce Records Standards*, Fluor Hanford. Richland, Washington, 2001.

HNF-SD-WM-QAPP-036, Revision 4, *Waste Management Project Quality Assurance Program Plan*, Fluor Hanford, Richland, Washington, 2001.

HNF-SD-WM-TI-794, Rev. 0, *Activity of Fuel Batches Processed Through Hanford Separations Plants, IY44 through 1989*, Lockheed Martin Hanford Corporation. Richland, Washington, 1997.

HW-10475-C, *Hanford Technical Manual, Section C, Separations*, Hanford Engineer Works, Richland. Washington, 1944.

WHC-MR-0452. *A Brief History of the T-Plant Facility, Hanford* Site, Westinghouse Hanford Company, Richland, Washington, 1994.

WHC-SD-WM-RPT-267, 1996, *Basis for Dose Rate to Curie Assay Method,* Westinghouse Hanford Company, Richland, Washington.

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Table 1. General Source Terms for Radioactive Material In T Plant Complex Waste Streams.

Isotope	Percentage of activity from representative sources		
	Fuel processed at T Plant Complex[1]	All Hanford Site fuel[1]	All tank waste[2]
H-3	0.050%	0.078%	0.018%
C-14	0.003%	0.002%	0.003%
Ni-59	<0.001%	0.001%	0.001%
Co-60	<0.001%	0.003%	0.014%
Ni-63	0.078%	0.054%	0.085%
Se-79	<0.001%	<0.001%	0.001%
Sr-90	44.592%	42.473%	48.740%
Nb-93m	0.002%	0.001%	0.002%
Zr-93	0.003%	0.002%	0.003%
Tc-99	0.022%	0.016%	0.029%
Ru-106	<0.001%	<0.001%	0.001%
Cd-113m	0.006%	0.008%	0.013%
Sb-125	<0.001%	0.018%	0.035%
Sn-126	0.001%	0.001%	0.001%
I-129	<0.001%	<0.001%	<0.001%
Cs-134	<0.001%	0.004%	0.008%
Cs-137	51.032%	49.531%	48.168%
Sm-151	0.238%	1.624%	2.327%
Eu-152	<0.001%	0.001%	0.003%
Eu-154	0.008%	0.048%	0.113%
Eu-155	0.058%	0.029%	0.099%
U-233	<0.001%	0.006%	<0.001%
U-234	0.050%	0.017%	0.003%
U-235	0.002%	0.001%	<0.001%
U-236	<0.001%	0.001%	<0.001%
Np-237	<0.001%	<0.001%	<0.001%
Pu-238	0.018%	0.059%	0.003%
U-238	0.051%	0.016%	<0.001%
Pu-239	2.827%	1.801%	0.055%
Pu-240	0.246%	0.328%	0.010%
Am-241	0.166%	0.428%	0.123%
Pu-241	0.547%	3.448%	0.140%
Cm-242	<0.001%	<0.001%	<0.001%
Pu-242	<0.001%	<0.001%	<0.001%
Am-243	<0.001%	<0.001%	<0.001%
Cm-243	<0.001%	<0.001%	<0.001%
Cm-244	<0.001%	<0.001%	<0.001%

Data sources:
[1]HNF-SD-WM-TI-794, decay corrected from 1994 to 2001
[2]TWINS database, current decayed data.

Table 2. Percentage of Limits for Bounding Contact-Handled Waste Based on Total Hanford Site Fuel Distribution.

Isotope	Percent of mobile radionuclide reporting limit	Percent of Category 1 limit	Percent of Category 3 limit	Percent of A2 value
H-3	0.006%	<0.001%	NL	<0.001%
C-14	5.847%	0.008%	<0.001%	<0.001%
Ni-59	NL	<0.001%	<0.001%	<0.001%
Co-60	NL	<0.001%	NL	<0.001%
Ni-63	NL	0.003%	<0.001%	<0.001%
Se-79	4.565%	<0.001%	<0.001%	<0.001%
Sr-90	NL	850.700%	<0.001%	31.759%
Zr-93	NL	<0.001%	<0.001%	0.001%
Tc-99	24.492%	0.224%	0.001%	0.001%
Ru-106	NL	NL	NL	<0.001%
Cd-113m	NL	0.003%	NL	0.006%
Sb-125	NL	NL	NL	0.002%
Sn-126	NL	1.486%	<0.001%	<0.001%
I-129	9.947%	0.001%	<0.001%	NL
Cs-134	NL	NL	NL	0.001%
Cs-137	NL	2886.003%	0.001%	7.407%
Sm-151	NL	0.011%	<0.001%	0.030%
Eu-152	NL	0.004%	NL	<0.001%
Eu-154	NL	0.020%	NL	0.007%
Eu-155	NL	NL	NL	0.001%
U-233	129.870%	0.246%	0.002%	0.424%
U-234	394.602%	0.621%	0.003%	1.289%
U-235	16.268%	0.081%	<0.001%	NL
U-236	13.007%	0.019%	<0.001%	0.042%
Np-237	3.889%	0.063%	<0.001%	0.050%
Pu-238	NL	4.050%	0.001%	22.166%
U-238	364.887%	0.896%	0.004%	NL
Pu-239	NL	303.811%	1.374%	672.202%
Pu-240	NL	55.368%	0.245%	122.505%
Am-241	NL	65.388%	0.162%	159.905%
Pu-241	NL	18.117%	0.044%	25.786%
Cm-242	NL	NL	NL	<0.001%
Pu-242	NL	0.003%	<0.001%	0.008%
Am-243	NL	0.007%	<0.001%	0.008%
Cm-243	NL	<0.001%	<0.001%	0.001%
Cm-244	NL	<0.001%	<0.001%	0.022%

NL = No applicable limit for the isotope.

Sr-90	Most significant fission product for transportation purposes.
Cs-137	Mosr significant isotope for waste category disposal limits.
Uranium isotopes (U-233, 234, 235, 236, and 238)	Could exceed the mobile radionuclide reporting limits; might not be a radionuclide of concern for transuranic waste. as onsite disposal limits would not apply.
Plutonium isotopes (Pu-238, 239. 240, and 241)	Transuranic radionuclides; significant for transportation and disposal.
Am-241	Transuranic radionuclide; significant for transportation and disposal.
Other possible radionuclides of interest:	
C-14	Could approach mobile radionuclide reporting limit only in highest activity waste streams.
Se-79	Could approach mobile radionuclide reporting limit only in highest activity waste streams.
Tc-99	Could approach mobile radionuclide reporting limit only in highest activity waste streams.
Sn-126	Could exceed 1% of the Category 1 limit in high activity waste streams.
I-129	Could approach mobile radionuclide reporting limit only in highest activity waste streams.
Sm-151	Might contribute more than 1% of the total activity mixture (not significant with respect to any specific limits).
Np-237	Could approach mobile radionuclide reporting limit only in highest activity waste streams.

Radionuclide	Analysis method	Interpretation of data
Sr-90	Sr-90 separation! beta analysis	None.
cs-137	Gamma energy analysis (GEA)	None.
Uranium	Isotopic uranium analysis (ICP/MS or equivalent)	None.
	Total uranium analysis – uranium phosphorescence	Isotopic distribution is determined from HNF-SD-WM-TI-794 or other appropriate data,.
Plutonium	Separation on resin/alpha energy analysis (AEA)	Isotopes not quantified by the method are determined from Pu distributions of HNF-SD-WM-TI-794 or other appropriate data.
	Total alpha analysis	Distribution of plutonium isotopes is determined from HNF-SD-WM-TI-794 or other appropriate data. Uranium alpha activity can be subtracted from the total alpha result.
An-241	GEA or AEA	None.
	Total alpha analysis	Activity of Am-241 is determined from HNF-SD-WM-TI-794 or other appropriate data,
Other high-energy gamma emitting isotopes (e.g., Co-60, Cs-134, Eu-152, Eu-154., Eu-155., Np-237)	These isotopes are not expected to he significant in the waste. hut can be detected by GEA.	None.
	counting (LSC)	
Sc-79	Separation/LSC	None.
Tc-9Y	Separation/LSC	None.
I-I29	Separation/LSC	None.

Table 5. Source Data for Process Knowledge-Based Scaling Factors.

Waste type	Data source
Bismuth phosphate process equipment and other waste from process cells that were not used for decontamination	HNF-SD-WM-TI-791, Table 2, total activity of fuel processed through T Plant Complex.
Tank Fanns equipment	Global tank inventory froin TWINS database.
Other equipment from decontamination	HNF-SD-WM-TI-794, Table 2, total activity of fuel processed through all separations buildings.

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APPENDIX A

SCALING FACTORS FOR T PLANT COMPLEX CANYON DECK LEGACY CONTAMINATION

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APPENDIX A

SCALING FACTORS FORT PLANT COMPLEX CANYON DECK LEGACY CONTAMINATION

Discussion of Available Data

In previous T Plant Complex characterization documents. scaling factors for legacy contamination on the 22 1-T Canyon Building deck and 211-T Facility tunnel were based largely on analyses from tanks 5-7, 6-1, and 15-1. In December of 2000. smear samples were obtained to determine the relative distribution of Sr-90, Cs-137, and plutonium isotopes in the canyon. The radiological control organization obtained 38 smear samples from representative locations within the canyon (sample numbers 22 IT-00-133 through 170). These samples were analyzed by total alpha. total beta. GEA, Sr-90, and isotopic Pu analyses.

Calculation of Scaling Factors

The scaling factors for the primary isotopes of concern (Sr-90, Pu-238, Pu-239, Pu-240, Pu-241, and Am-241) were determined from the December 2000 samples. These scaling factors were calculated as a weighted average of the analysis results. A weighted average of the analysis results was chosen over a simple average of the scaling factors because this average **is** more representative of the areas of highest contamination (correlated with the most containinated smear samples) and minimizes the effect of nondetects among the sample results. Calculation of the Sr-90/Cs-137 and Pu-239+240/Cs-137 scaling factors are provided in Table A-1

Table **A-2** shows the complete set of scaling factors for T Plant Complex Canyon Building deck waste. Scaling factors for isotopes other than Sr-90, Pu-238, Pu-239, Pu-240. Pu-241, and Am-241 were derived directly from HNF-SD-WM-TI-791, Table 2, using the Total Fuel column (i.e., the total fuel from Hanford Site reactors processed through Hanford Site separations facilities). The specific Pu-238, Pit-239. Pu-240, Pit-24 1. and Am-241 scaling factors were calculated using the isotopic plutonium and Ani-241 distributions froin HNF-SD-WM-TI-794, Table *2*.

Based on historical data and comparison of total alpha and plutonium alpha results from the 2000 samples. uranium analysis is not necessary. It is highly unlikely that waste from uranium separations is present at T Plant Complex.

Table A-2 shows that the new Sr-90/Cs-137 scaling factor is slightly lower than the previous factor. At the same time, the scaling factors for Pit-239 and Pu-240 increased by roughly a factor of 10. while the Pu-241 scaling factor decreased by a factor of 2. The Am-241/Cs-137 scaling factor increased slightly.

Other isotopes in Table A-2 previously did not have scaling factors. In most cases, these isotopes will not be significant for disposal or U.S. Department of Transportation purposes. However. these isotopes have been calculated so that the determination whether or not an isotope is reportable can he computed on a container-by-container basis.

Table A-1. Calculation of Sr-90 and Pu-239/240 to Cs-137 Scaling Factors from December 2000 Data.

	Sample results			Sample scaling factors		
Sample #	Sr-90 (uCi)	Cs-137 (uCi)	Pu-239/240 (uCi)	Sr-90 to Cs-137	Pu-239/240 to Cs-137	Notes
221T-00-0133	1.12E-03	6.55E-04	8.72E-06	1.71E+00	1.33E-02	
221T-00-0134	7.40E-04	5.59E-05	5.40E-06	1.32E+01	9.66E-02	Pu below detection limit
221T-00-0135	6.39E-04	1.42E-03	5.81E-06	4.50E-01	4.09E-03	
221T-00-0136	3.86E-05	3.34E-05	5.26E-06	1.16E+00	1.57E-01	Cs & Pu below detection limit
221T-00-0137	5.18E-04	7.33E-05	8.11E-06	7.07E+00	1.11E-01	
221T-00-0138	6.03E-04	2.94E-04	8.82E-06	2.05E+00	3.00E-02	
221T-00-0139	2.26E-03	2.35E-03	3.13E-05	9.62E-01	1.33E-02	
221T-00-0140	1.99E-03	4.30E-03	5.84E-05	4.63E-01	1.36E-02	
221T-00-0141	2.29E-03	1.95E-04	6.83E-06	1.17E+01	3.50E-02	
221T-00-0142	1.24E-02	9.82E-04	2.68E-04	1.26E+01	2.73E-01	
221T-00-0143	3.38E-03	2.12E-03	5.36E-04	1.59E+00	2.53E-01	
221T-00-0144	6.69E-04	1.04E-03	1.93E-04	6.43E-01	1.86E-01	
221T-00-0145	5.63E-03	1.99E-03	3.63E-05	2.83E+00	1.82E-02	
221T-00-0146	3.26E-03	9.45E-04	2.38E-05	3.45E+00	2.52E-02	
221T-00-0147	2.57E-03	2.87E-03	3.28E-05	8.95E-01	1.14E-02	
221T-00-0148	5.96E-03	9.25E-04	4.41E-05	6.44E+00	4.77E-02	
221T-00-0149	2.61E-03	1.10E-03	1.97E-05	2.37E+00	1.79E-02	
221T-00-0150	5.22E-03	2.17E-03	7.21E-05	2.41E+00	3.32E-02	
221T-00-0151	1.09E-03	5.98E-04	2.57E-05	1.82E+00	4.30E-02	
221T-00-0152	3.40E-03	8.04E-04	6.91E-05	4.23E+00	8.59E-02	
221T-00-0153	6.18E-03	1.42E-03	3.35E-04	4.35E+00	2.36E-01	
221T-00-0154	1.42E-03	5.94E-04	5.52E-05	2.39E+00	9.29E-02	
221T-00-0155	3.20E-03	3.29E-04	8.13E-06	9.73E+00	2.47E-02	
221T-00-0156	9.25E-04	1.07E-03	2.00E-05	8.64E-01	1.87E-02	
221T-00-0157	6.25E-04	2.02E-04	1.07E-05	3.09E+00	5.30E-02	
221T-00-0158	1.56E-03	2.39E-03	1.70E-04	6.53E-01	7.11E-02	
221T-00-0159	3.98E-03	1.26E-03	7.76E-05	3.16E+00	6.16E-02	
221T-00-0160	2.35E-03	3.96E-04	1.95E-05	5.93E+00	4.92E-02	
221T-00-0161	8.42E-03	8.72E-04	5.66E-05	9.66E+00	6.49E-02	
221T-00-0162	3.85E-03	6.74E-04	2.67E-05	5.71E+00	3.96E-02	
221T-00-0163	2.87E-03	7.12E-04	2.16E-05	4.03E+00	3.03E-02	
221T-00-0164	3.44E-03	2.96E-04	8.17E-05	1.16E+01	2.76E-01	
221T-00-0165	3.66E-03	3.11E-04	2.33E-05	1.18E+01	7.49E-02	
221T-00-0166	5.69E-03	6.68E-04	4.23E-05	8.52E+00	6.33E-02	
221T-00-0167	2.05E-03	7.21E-04	4.03E-05	2.84E+00	5.59E-02	
221T-00-0168	4.74E-03	4.53E-04	4.27E-05	1.05E+01	9.43E-02	
221T-00-0169	2.15E-03	6.67E-04	1.69E-05	3.22E+00	2.53E-02	
221T-00-0170	6.72E-03	4.91E-04	1.55E-04	1.37E+01	3.16E-01	
Sum	1.20E-01	3.84E-02	2.66E-03			
Ratio to Cs-137[1]	3.13E+00	1.00E+00	6.93E-02			

[1] Calculated as a weighted average.

Table A-2. Scaling Factors for T Plant Complex Canyon Building Deck Legacy Contamination.

Isotope	Previous scaling factor	New scaling factor	Calculation basis
C-14	None	4.79E-05	1
Se-79	None	9.78E-06	1
Sr-90	4.71	3.13	2
Tc-99	None	3.24E-04	1
Sn-126	None	1.50E-05	1
I-129	None	6.27E-07	1
Cs-137	1.00E+00	1.00E+00	NA – reference isotope
Sm-151	None	3.28E-02	1
U-233	None	1.15E-04	1
U-234	None	3.48E-04	1
U-235	None	1.44E-05	1
U-236	None	1.15E-05	1
U-238	None	3.22E-04	1
Np-237	None	2.70E-06	1
Pu-238	None	1.93E-03	3
Pu-239	5.45E-03	5.86E-02	3
Po-240	1.33E-03	1.07E-02	**3**
Am-241	1.12E-02	1.39E-02	3
Pu-241	2.12E-01	1.12E-01	**3**

1. Scaling factors for C-14, Se-79, Tc-99, Sn-126, I-129, Sm-151, uranium isotopes, and Np-237 were derived directly from HNF-SD-WM-TI-794. Table 2, Total Fuel column by dividing the activity of each respective isotope by the activity of Cs-137, decay corrected from 1994 to 2001.
2. The scaling factor for Sr-90 is the weighted average of Sr-90 to Cs-137 results from the **T** Plant Complex Canyon Building characterization sampling performed in December 2000.
3. The Pu-238, Pu-239, Pu-240, Pu-241, and Am-241 scaling factors were derived using the weighted average of Pu-239/240 results to Cs-137 results from the T Plant Complex Canyon Building characterization sampling performed in December 2000. The alpha activity was apportioned to Pu-238, Pu-239, Pu-240, Pu-241, and Am-241 according to HNF-SD-WM-TI-794, Table **2,** Total Fuel column (i.e., activity of each isotope divided by the sum of the Pu-239 and Pu-240 activity), decay corrected from 1994 to 2001.

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APPENDIX B

SCALING FACTORS FOR PROCESS CELL WASTE

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APPENDIX B

SCALING FACTORS FOR PROCESS CELL WASTE

Discussion of Available Data

Relevant data for process cell waste extremely are limited. Sampling and analysis or **NDA** will be performed on this waste stream to determine TRU scaling factors. The Sr-90 scaling factor will be confirmed by sampling and analysis.

Calculation of Scaling Factors

The scaling factors in Table B-I are derived from HNF-SD-WM-TI-794. TRU scaling factors will be developed on a case-by-case basis from analysis data and/or NDA.

Table B-1. Scaling Factors for Non-Transuranic Radionculides in Process Cell Waste.

Isotope	Original process cells[1]	Decontamination cells[2]
C-14	6.24E-05	4.79E-05
Se-79	1.32E-05	9.78E-06
Sr-90	8.74E-01	8.58E-01
Tc-99	4.33E-04	3.24E-04
Sn-126	1.98E-05	1.50E-05
I-129	8.16E-07	6.27E-07
Cs-137	1.00E+00	1.00E+00
Sm-151	4.67E-03	3.28E-02
U-233	8.98E-10	1.15E-04
U-234	9.76E-04	3.48E-04
U-235	4.34E-05	1.44E-05
U-236	8.35E-06	1.15E-05
U-238	3.54E-04	3.22E-04
Np-237	2.67E-06	2.70E-06

[1] Derived from HNF-SD-WM-TI-794, Table 2. T Plant Complex Fuel column by dividing the activity of each respective isotope by the activity of Cs-137, decay corrected from 1994 to 2001.

[2] Derived from HNF-SD-WM-TI-794, Table 2, Total Hanford Fuel column by dividing the activity of each respective isotope by the activity of Cs-137, decay corrected from 1994 to 2001

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