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Assessment of Physical Properties of Transuranic Waste in Hanford Single-Shell Tanks

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March 2003

Prepared for the U.S. Department of Energy
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Richland, WA 99352

Executive Summary

CH2M HILL Hanford Group, Inc. (CH2M HILL) is identifying and developing supplemental process technologies to accelerate the tank waste cleanup mission at Hanford. One such technology targets the disposal of Hanford transuranic (TRU) process wastes stored in single-shell tanks (SSTs). Ten Hanford SSTs are candidates for designation as contact-handled TRU waste: the B-200 series tanks (B-201, B-202, B-203, and B-204), the T-200 series tanks (T-201, T-202, T-203, and -T-204), plus T-110 and T-111.[a] The retrieval of these tanks is intended to be a "dry" process in which these wastes will be retrieved from the tanks *as is* or with a recycled liquid stream to help mobilize the waste in the tank and through transfer lines and vessels. Subsequently, the retrieved waste will be dewatered to remove excess liquid and transferred to waste packages in a form suitable for disposal at the Waste Isolation Pilot Plant (WIPP).

An understanding of waste physical properties is needed to support design and procurement of the SST TRU handling and packaging system and to produce suitable physical simulants to test such a process. Pacific Northwest National Laboratory (PNNL) has been tasked with developing these waste simulants. This report summarizes PNNL's assessment of available waste physical property information for the 10 candidate TRU SSTs. Data sources include the Hanford Tank Waste Information Network System (TWINS) database, technical reports, and visual observations from reviewing photographs and videotape recordings taken during the extrusion of various SST TRU waste core samples.

While the retrieval process is expected to alter certain waste physical properties such as shear strength, the effects of this process on waste properties cannot yet be quantified. Therefore, the scope of this report is to describe the properties of SST TRU wastes as they are known for unprocessed wastes or, in some cases, for diluted waste samples. The report focuses on the rheology, particle properties, settling characteristics, bulk density, and water content of the waste. Because some physical properties of the waste are related to the chemistry of the waste (e.g., particle hardness), a brief overview is provided of the process and chemistry resulting in the TRU waste. Other properties, including the potential for gas retention in the waste, are discussed briefly. Qualitative descriptions of how waste properties might be affected by retrieval and subsequent processing are also noted.

To provide a broad understanding of the waste rheology in all the TRU SSTs, photographs and videotapes of core extrusions were analyzed for seven of the ten tanks (B-203, B-204, the T-200 series, and T-110). Typically, waste ranged from very wet near the waste surface to a ductile, moderate-strength sludge below the surface, progressing to more brittle and stronger waste toward the bottom of the tank. Methodologies were developed to use extrusion length and

(a) Hanford waste tanks are designated with the prefix 241-. In this report, as in common usage, the prefix is omitted.

slump measurements obtained from the videotapes to estimate the waste shear strength using previously reported extrusion data for ductile (bentonite clay/water) and brittle (kaolin clay/colloidal silica/water) simulants as reference. Applying these methods, the shear strength of the sludge in the seven tanks was estimated to range from <30 to ~4000 Pa. The bulk of the waste was categorized as ductile solid to moderately brittle with a shear strength range of 200 to 2000 Pa. Observed waste rheology was similar across the set of TRU SSTs. The only notable difference was that the T-110 waste was somewhat more ductile up to higher shear strengths.

Limited amounts of rheology data have been previously reported for B-201, B-202, and T-111 waste samples, and these shear strength and viscosity data are discussed in this report. Measured shear strengths for six samples from B-201 and B-202 ranged from 200 to 1410 Pa, consistent with the shear strengths estimated from core extrusions of other B- and T-200 series wastes. Similarly, the shear strength of undiluted T-111 samples was 500 ± 230 Pa. Viscosity results for B-201 and B-202 waste samples diluted 1:1 (volume basis) with water are reported in the form of yield power-law function parameters and shown graphically. At a 100 s^{-1} strain rate, the viscosity of the diluted waste ranged from 5 to 15 cP for B-201 and from 2.1 to 4.5 cP for B-202. Diluted T-111 samples also exhibited pseudoplastic behavior, but results were not correlated to yield power-law functions because the viscosity was near the system detection limit (2 cP).

Particle size and solids settling data are of value for developing waste dewatering and handling processes. Particle size distribution results from earlier studies of B-201 and T-111 waste samples are summarized, and volume basis size distribution plots are shown for B-201. While the mean density particle size is consistently <2 μm for both tank wastes, the mean particle size in terms of volume fraction ranges from ~7 to 66 μm for segments taken from two core samples of B-201 and from ~8 to 65 μm for T-111 samples. There were no apparent trends in particle size distribution with vertical location in the tank.

Gravitational and centrifugal settling data for as-received (undiluted) and water-diluted (1:1 and 3:1 volume basis) samples of B-201, B-202, and T-111 wastes are presented. Essentially no settling occurred in the small undiluted samples at 1 G (gravitational force on Earth), whereas centrifugation at more than 1000 G produced 2 to 12 vol% free liquid in B-201 samples, 30 to 40 vol% free liquid in B-202 samples, and 28 to 34 vol% free liquid in T-111 samples. The rate of gravity settling in diluted samples decreased significantly after two days, but centrifugation clearly indicates that settling was not complete. For example, gravity settling for two days produced free liquid of 8 to 19 vol% in 1:1 (by volume) diluted B-201 samples, whereas centrifugal settling increased the free liquid to 42 to 58 vol%.

The estimated shear strength of the wastes generally increased with depth in the tank, possibly because of waste self-compaction due to lithostatic loading. The water content (and bulk density) of the waste also appears to vary with location in the tank, with lower moisture content near the tank bottom. On the whole, the water content of waste sludge was found to

range from 70 to 85 wt%; the reported water content is somewhat higher in Tank T-111 (85 to 90 wt%) and lower in a few samples from T-201 (~65 wt%). Most of the data for bulk solids samples, a matrix of waste solids and interstitial liquid, show bulk densities of 1.15 to 1.30 g/mL, and the density generally increases with decreasing water content. The shear strength estimates obtained from the extrusion methods were compared with the water content and bulk density of waste samples from the same core segments. The shear strength and, to a lesser extent, the density show some tendency to decrease with increasing water, but significant scatter exists in the data.

The physical properties of in situ and diluted SST TRU waste described in this report and summarized in the discussion above are tabulated in Table ES.1. In many cases, the expected range of properties is estimated from limited data. However, in those instances where data are available for many tanks and multiple locations within tanks, the data do not indicate major differences among individual tanks. Therefore, it appears reasonable to treat individual tank results as typical of Hanford SST TRU waste.

Table S.1. Expected Range of Physical Properties of In Situ and Diluted SST TRU Waste

Property	Expected Range	Comments
Shear strength	200 to 2,000 Pa (majority of waste) 0 to 4,000 Pa (range, including liquid)	Estimated from data obtained from core extrusions (Sections 3.1 and 3.2) and reported shear vane measurements (Section 4.1).
Viscosity	2 to 25 cP at 10 s^{-1} 2 to 15 cP at 100 s^{-1}	Results for 1:1 dilution with water; higher viscosities expected for waste diluted less and at lower strain rates. Waste exhibited pseudoplastic rheology. Section 4.1.
Waste settling	~0 vol% free liquid (1 G) 2 to 40 vol% free liquid (>1000 G, 1 hr)	Undiluted waste, >200 Pa shear strength. Weaker waste (liquid in the extreme) is expected to produce more free liquid on settling. Section 4.2.
Waste settling	5 to 25 vol% free liquid (1 G, ~2 days) 40 to 60 vol% free liquid (>1000 G, 1 hr)	1:1 diluted waste, >200 Pa shear strength prior to dilution. See comment above. Section 4.2.
Waste settling	40 to 65 vol% free liquid (1 G, ~2 days) 70 to 85 vol% free liquid (>1000 G, 1 hr)	3:1 diluted waste, >200 Pa shear strength prior to dilution. See comment above. Section 4.2.
Particle size, mean	7 to 70 μm (volume density) < 2 μm (number density)	Section 4.3.
Water content	70 to 85 wt% (majority of waste) 65 to 90 wt% (range)	Section 5.1.
Liquid Density	~1.05 g/mL	Section 5.2
Bulk Density	1.15 to 1.3 g/mL (majority of waste) 1.1 to 1.4 g/mL (range)	Section 5.2

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1.0 Introduction

CH2M HILL Hanford Group, Inc. (CH2M HILL) is in the process of identifying and developing supplemental process technologies to accelerate the tank waste cleanup mission. A range of technologies is being evaluated to allow disposal of Hanford waste types, including transuranic (TRU) process wastes. Gasper et al. (2002) identified 12 Hanford waste tanks that may meet the criteria for designation as TRU waste, including three double-shell tanks (AW-103, AW-105, and SY-102)[a] and nine single-shell tanks (SSTs): the B-200 series (B-201, B-202, B-203, and B-204), the T-200 series (T-201, T-202, T-203, and T-204), and Tank T-111. The SST T-110 has recently received attention as a candidate for designation as a TRU waste type. If not designated as TRU waste, it might be defined as low-level waste that could be handled and packaged with the same process used for other contact-handled TRU (CH-TRU) wastes from the SSTs.

Originally, a "dry" waste retrieval process was considered for these tanks—that is, no liquid was to be used to slurry, or soften, the waste for the process of retrieving and transferring the waste to a packaging system. In such a retrieval and transfer process, air (or other process gas) would be forced into the in-tank waste to decrease its effective density so that the "fluffed" waste could be vacuumed out of the tank, and the removed waste would accumulate in a hopper, with the bulk sludge (possibly still moist and having some entrained trapped gas) being transferred from the hopper into storage drums. The current retrieval plans call for a modified dry retrieval process in which a recycled liquid stream (minimal fresh water) flowing at ~1.4 to 5 gpm is used to help mobilize the waste in the tank and through transfer lines and vessels. This retrieval approach requires that a significant portion of the liquid be removed from the mobilized waste sludge in a "dewatering" process such as gravity settling, centrifugation, or drying prior to transferring it to waste packages. Additionally, adsorbent may be added to the waste packages (e.g., drums) to prevent formation of a free liquid layer.

In support of CH2M HILL's effort to develop a TRU waste handling and packaging process, Pacific Northwest National Laboratory (PNNL) has been tasked with developing waste simulants. For the SST CH-TRU wastes, the suite of simulants is likely to include a nonradioactive chemical simulant of the waste liquid that will potentially be separated from the waste solids. This simulant would be used to evaluate packaging and sorbent material compatibility. Simulants that reproduce the important physical behavior of the waste, including mechanical, flow, and dewatering properties, are also needed to develop and evaluate the TRU waste handling and packaging process. To produce suitable physical simulants, PNNL is now evaluating TRU waste physical properties. This report summarizes PNNL's assessment of available physical property information for the 10 candidate TRU SSTs, including T-110.

(a) Hanford waste tanks are designated with the prefix 241-. In this report, as in common usage, the prefix is omitted.

The information provided in this report will be supplemented in the near future with chemical and physical property data obtained from actual waste sample composites of Tanks B-203, T-203, T-204, and T-110. Experimental studies on these samples are ongoing at Hanford. It is expected that the tests will be completed and the results reported in fiscal year 2003.

The effects of the proposed CH-TRU retrieval and transfer systems on waste properties have not been quantified. Accordingly, the scope of this report is to describe the properties of TRU wastes as they are known for unprocessed wastes or, in some cases, for diluted waste samples. Qualitative descriptions of how waste properties might be affected by retrieval and transport are noted where appropriate.

This report focuses on the waste rheology, settling characteristics, particle properties, bulk density, and water content. Because some physical properties of the waste are related to the waste chemistry (e.g., particle hardness), a brief overview of the process and chemistry resulting in the TRU waste is provided in Section 2. Section 3 shows representative photographs of extruded SST TRU wastes, describes visual observations, and summarizes estimates of the waste shear strength derived from videotaped core extrusions. Other waste strength and viscosity measurements, gravity and centrifugal settling data, and particle properties, including size distribution, are discussed in Section 4. Section 5 summarizes the variation in water content and bulk density of the waste in the tanks and addresses the relationship of these properties and waste shear strength. Other properties, including the potential for gas retention in the waste, are discussed briefly in Section 6. Cited references are listed in Section 7, and the extrusion length and slump methodologies are detailed in the appendix. Data sources for this report include the Hanford Tank Waste Information Network System (TWINS) database, technical reports, and visual observations from the review of photographs and videotape recordings taken during the extrusion of various tank waste core samples.

2.0 Overview of the Waste Chemistry

Although the physical simulants being developed are not expected to have the same chemical composition as the actual waste, this section of the report provides some background on the waste sources and composition for general information. Knowledge of the history and composition of the waste supplements the specific physical property data in the rest of this report.

2.1 History of TRU Waste Transfer

The wastes contained in the B-200 series, the T-200 series, T-110, and T-111 tanks originated primarily in the 224 Building waste stream, which was derived from the lanthanum fluoride decontamination step in the bismuth phosphate process (e.g., Gasper et al. 2002).

The tanks in the B-200 and T-200 series are considered to contain only 224 Building wastes (224 waste), based on waste transfer records. Tank B-201 received waste from the 224-B Concentration building from October 2, 1946 through October 1948, after which the tank was considered filled with solids and the 224-B Concentration building waste was diverted to Tank B-204, which was connected in a cascade with Tanks B-203 and B-202. Liquid was gravity discharged from the last tank in the cascade, B-202, to the 241-B-1 and 241-B-2 cribs. Solids contained in the 224-B Building waste were allowed to settle in Tanks B-204 through B-202. The cascade of Tanks B-204, B-203, and B-202 continued to receive 224-B Concentration building wastes until September 1952. The history of the filling of T-200 series tanks parallels that of the B-200 series tanks. Tank T-201 received waste from the 224-T Concentration building from November 4, 1946 through May 24, 1949, after which the tank was considered filled with solids and the 224-T Concentration building waste was routed to Tank T-204. Tank T-204 was connected in a cascade with Tanks T-203 and T-202. Liquid was gravity discharged from the last tank in the cascade, T-202, to the 241-T-1 and 241-T-2 cribs. Solids contained in the 224-T Building waste were allowed to settle in T-204 through T-202.[(a)]

Some 224 waste is also found in Tanks T-110 and T-111, along with "2C" waste, which is second-cycle decontamination waste from the bismuth phosphate process. Tanks T-110, T-111, and T-112 initially operated in a cascade, with waste received into the lead tank and allowed to overflow to the next tank in series. Tank T-110 received only 2C waste from January 1945 through May 1952. From May 1952 through December 1954, Tank T-110 received a mixture of 2C and 224 wastes. Waste was received into Tank T-110 and allowed to overflow into Tank T-111 until December 1954, when the primary waste stream was directed into T-111 and T-110 stopped receiving waste. Tank T-111 received only 2C waste that cascaded from Tank T-110 from January 1945 through May 1952. From May 1952 through October 1956, Tank T-111

(a) Johnson, ME. January 2003. *Origin of Wastes in the B-200 and T-200 Series Single-Shell Tanks.* RPP-13300, draft, CH2M HILL Hanford Group, Richland, WA.

received a mixture of 2C and 224 wastes. From February 1960 through June 1967, dilute liquid equipment decontamination waste from T-plant was fed to Tank T-111.[a]

2.2 TRU Waste Composition

In the SST CH-TRU waste being considered for dry retrieval, bismuth (Bi) and sodium (Na) are the major metallic analytes, and nitrate (NO_3) is the major anion (Lumetta and Rapko 1994). Table 2.1 gives the bulk concentrations (including both solid and liquid phases) of the major constituents for nine of the 10 tanks; there is no bulk concentration data available for T-110. Water, which makes up most of the waste mass, is not included as a constituent in the table. The concentrations given in the table are based on core samples taken between 1991 and 1997. (Core sampling, in effect, forces a 1.125-in. inside-diameter pipe down through the waste and collects samples, each one a 19-in.-long segment, over the entire depth of the waste.)

Table 2.1. Waste Composition Ranges (major analytes)

Tank	Bi (wt%)	Na (wt%)	NO_3 (wt%)	Others (wt%)	
B-201	8.7 – 12	2.9 – 6.6	4.8 – 5.5	Si, 0.6 – 6.1 Mn, 1.5 – 3.2 Fe, 0.6 – 2.3 Ca, 0.5 – 2.3 La, 0.9 – 1.7	K, 0.3 – 1.6 Al, 0.1 – 1.4 B, < 0.1 – 1.5 P, 0.4 – 0.8 F, 0.6
B-202	1.4 – 5.0	3.4 – 6.5	5.1 – 7.1	Fe, 0.3 – 6.6 La, 0.7 – 2.6 Mn, 0.5 – 2.6 Ca, < 0.1 – 2.6 K, 0.6 – 1.6	Al, < 0.1 – 1.4 P, 0.2 – 1.2 F, 0.6 Si, 0.1 – 0.6 B, < 0.1
B-203[a,b]	1.4 – 5.6	2.6 – 3.2	3.0 – 5.6	Mn, 0.5 – 2.4 La, 0.8 – 1.5 Fe, 0.2 – 0.8 F, 0.6 – 0.7 K, 0.4 – 0.5	P, 0.2 Si, < 0.1 – 0.1 Ca, < 0.1 Al, < 0.1 B, < 0.1
B-204[a]	3.6 – 6.3	2.3 – 2.9	3.6 – 5.6	Mn, 0.9 – 1.8 La, 0.8 – 1.4 Fe, 0.2 – 1.1 F, 0.5 – 0.7 K, 0.5 – 0.7	P, 0.2 – 0.3 Si, < 0.1 – 0.2 Ca, < 0.1 Al, < 0.1 B, < 0.1

(a) Johnson, ME. January 2003. *Origin and Classification of Wastes in Single-Shell Tanks 241-T-110 and 241-T-111.* RPP-13873, draft, CH2M HILL Hanford Group, Richland, WA.

Table 2.1 (contd)

Tank	Bi (wt%)	Na (wt%)	NO_3 (wt%)	Others (wt%)	
T-201	2.0 - 12	2.3 – 3.2	4.1 – 4.8	Mn, 0.7 – 4.4 La, 0.6 – 2.5 Fe, 0.4 – 0.9 F, 0.4 – 0.5 P, 0.1 – 0.5	K, 0.1 – 0.5 Si, 0.1 – 0.2 Ca, 0.1 Al, 0.1 B, < 0.1
T-202[a]	2.8 – 4.4	3.2 – 3.8	5.7 – 6.6	Fe, 0.2 – 2.4 Mn, 0.9 – 1.5 La, 0.9 – 1.3 K, 0.6 – 0.8 F, 0.6 – 0.7	P, 0.2 – 0.3 Si, 0.1 – 0.2 Ca, < 0.1 B, < 0.1 Al, < 0.1
T-203	2.6 – 6.2	3.2 – 4.5	4.3 – 7.4	Fe, 0.2 – 2.2 Mn, 0.5 – 1.6 La, 0.7 – 1.4 K, 0.7 F, 0.5 – 0.7	P, 0.2 – < 0.4 Si, < 0.1 – 0.2 Al, < 0.1 – 0.2 Ca, < 0.1 B, < 0.1
T-204[a, c]	5.2	3.2	5.5	Mn, 1.4 La, 1.2 K, 0.6 F, 0.6 Fe, 0.4	P, 0.3 Si, 0.2 Ca, < 0.1 Al, < 0.1 B, < 0.1
T-110	Only the liquid from this tank has been analyzed; there are no data for the bulk waste.				
T-111	0.1 – 3.5	2.2 – 4.1	3.7 – 5.8	Mn, 0.3 – 2.4 Fe, 1.2 – 2.0 P, 0.4 – 1.7 Si, 0.5 – 0.6 La, 0.3 – 0.5	Ca, 0.2 – 0.5 F, 0.1 – 0.5 K, 0.1 – 0.2 Al, 0.1 B, < 0.1

(a) In this tank, acid dissolution was used to prepare all samples for analysis. This form of preparation could be seen to cause considerable underestimation of many constituents in this type of waste in other tanks where other preparation procedures were used and a comparison was possible. Thus, analyte concentrations that are low in this tank, compared with other tanks not noted in this way, may be artifacts.

(b) One half-segment of a core sample, 120:10-LH, contained 10 times as much NO_3, F, and other anions as other segments and is ignored in this table as erroneous.

(c) In this tank analyses were made only on a composite of the entire core (a mixture of all the segments), not on individual segments. The average composition is available, but not a range.

3.0 Visual Observations and Shear Strength Estimates

The shear strength for all but three of the 10 TRU SSTs was estimated from videotape recordings of core sample extrusions. In this section, representative photos of extruded waste and estimates of strength derived from measurements of waste extrusion length and slump are provided. A potential alternative data source for estimating waste strength, load data taken during core sampling of the waste, is also discussed briefly in this section.

The strength estimates described below are for minimally disturbed material representative of the in-tank (in situ) condition. The process of retrieving waste from the tanks, with or without the aid of a liquid diluent, will undoubtedly disturb and likely weaken the waste that is transported to and staged in a vessel (e.g., hopper) for subsequent dewatering and packaging. Onishi et al. (2003a), evaluating pump mobilization of Hanford double-shell tank waste, concluded that prior waste disturbance would reduce the waste yield strength and that the recovery of strength is time-dependent. The effects on waste properties of the disturbance resulting from TRU SST waste retrieval are also likely time-dependent and difficult to quantify.

Although we do not know the strength and other rheological properties of the retrieved and processed TRU waste, it is reasonable to use physical property information from relatively undisturbed waste to help bound the range of expected behavior. It is also possible that the disturbed waste will substantially regain its original strength depending on how it is handled. For example, a centrifugal dewatering process could result in compacted waste similar in strength to that present in the waste tanks.

3.1 Visual Observations and Summary of Waste Shear Strength

Videotapes or photographs of core extrusions are available for the SST TRU wastes except for Tanks B-201, B-202, and T-111, which predate the use of photography during waste extrusions.[(a)] The similarities in process history surrounding the filling of these tanks suggests they have properties similar to their sister tanks (e.g., the B-200 and T-200 series tanks for B-201 and B-202, and T-110 for T-111). Visually, significant similarity exists among all the B-200 and T-200 series videos and photographs. This is not to say that they are homogeneous; rather, they fit substantially within a range of characteristics that may be found in any individual tank. The visually similar properties among the B-200 and T-200 series tank photos include color (dark brown to black), apparent moisture content (high sheen to dull), extrusion shape and fracturing (ranging from ductile to nearly brittle), and apparent strength. The T-110 waste is mustard yellow and is generally more ductile than the other observed waste samples. However, the range of apparent strength is bracketed by that estimated for the B-200 and T-200 series samples.

(a) Personal communication on December 2, 2002 from Ray Akita, Fluor Hanford, who also provided copies of the videotapes and photos.

Figure 3.1 shows photos representative of extruded core segments (nominally 19 in. length and 1.125 in. diameter) for core 122[a] from Tank B-203. Starting from the top of the figure, which corresponds to higher in the waste tank, extrusion segments #3 through #14 are shown. These photos were taken in Hanford hot cells in late December 1995 and early January 1996. Segment #3 has a pool of dark liquid[b] extruded ahead of weak, wet sludge. (The waste in segments #1 and #2, above #3, consisted almost entirely of liquid that is hard to see in the extrusion footage.) Segments #6 and #7 are relatively ductile, but segment #6 appears somewhat stronger and shows a series of fractures. The observations for segments #6 and #7 suggest that waste lower (deeper) in the tank (#7) can be weaker than that layered above it. In our assessment of the visual evidence, this is not routine but also not uncommon. The lower B-203 waste segments (>#8) in Figure 3.1 appear even stronger, drier, and more brittle than those layered above.

Note also that the waste characteristics can vary within segments. For example, the lower portion of B-203 segment #8, shown on the left side of Figure 3.1, is fractured and somewhat brittle, whereas the upper portion of the segment (right side of the photo near the piston) is ductile and weaker, like the bottom of segment #7 above it. This is likely due to vertical heterogeneity within the tank waste, but it could result in part from changes in the waste samples after core sampling and prior to extrusion. Core extrusions such as that depicted in Figure 3.1 provide a vertical picture of waste variation within a tank. Waste properties can also vary axially within tank waste.[c]

Figure 3.2 shows captured video images from in-process extrusions for segments from the B-200 and T-200 series tanks. The Figure 3.2 photos are ordered vertically from weakest (upper) to strongest (lower) and bound the range of behavior observed in Tank B-203 segments in Figure 3.1. For example, the upper photo in Figure 3.2 for Tank T-204 core 188 segment #3 shows weak, wet sludge as it is extruded, exhibiting visual characteristics similar to those observed in B-203 segment #3. The lower photo in Figure 3.2 for Tank T-201 core 192 segment #7 depicts relatively brittle and long subsegments characteristic of stronger waste fractions in the TRU SSTs.

The middle photos in Figure 3.2 were taken from the extrusion of Tank B-204 core 114 and are representative of the bulk of waste observed in the TRU SST videotapes and photos. Segment #4 (upper-middle photo) is continuous and relatively ductile, having a characteristic sigmoid extrusion shape. Segment #10 (lower-middle photo) is moderate strength, and it fractures, creating short to medium-length subsegments.

(a) The core number is sequential for cores taken from all Hanford waste tanks; it is not the number of cores taken from an individual tank.

(b) The Hanford TWINS database on December 12, 2002 indicated that four of the 10 SST TRU tanks (B-203, B-204, T-201, and T-110) have small amounts of residual supernatant liquid at the surface.

(c) For example, PA Meyer and WL Kuhn assess the potential redistribution of solids in a Hanford saltcake waste tank in letter report TWS02.074, entitled *Modeling Solids Redistribution in Tank S-112*, dated September 2002.

With other properties such as waste density and particle structure equal, an increasing length of extruded segments at failure (e.g., fracture for more brittle materials) is an indicator of increasing waste strength (Section 3.2 and the appendix). Many of the waste segments observed in the TRU SSTs had fractured subsegments ranging in length to the extremes depicted in the lower two photos of Figure 3.2.

Using the videotapes, we systematically assessed the shear strengths of the TRU sludge segments shown in Figure 3.2 from the measured extrusion length and slump discussed in Section 3.2. The method is an extension of the visually based core-extrusion shear strength estimation technique described by Gauglitz and Aikin (1997). They videotaped extrusions of bentonite clay/water and kaolin clay/colloidal silica (Ludox®)/water simulants of known and varying shear strengths and used the photographic results as a guide to estimating the shear strength of wastes exhibiting similar characteristics. Comparing their results with available tank waste strength data, they concluded that estimates of shear strength from horizontal extrusions were likely accurate within a factor of 2.

In Section 3.2 and the appendix to this report, we describe the horizontal extrusion length methodology that was applied here. The accuracy of the technique is not known, but a basis is given for placing bounds on the shear strength estimates that are approximately ± 25% of the central value. Table 3.1 summarizes the estimated ranges of shear strength resulting from the application of the extrusion length and slump techniques to the extrusion segments shown in Figure 3.2 and other comparable extruded waste segments. In the table we attempt to categorize visual observations of the SST TRU wastes and their typical range of estimated shear strengths. However, the visual descriptors do not correspond universally to the strength range indicated in Table 3.1. For example, ductile materials were observed in some cases (e.g., Tank T-110) with strengths greater than 200–700 Pa.

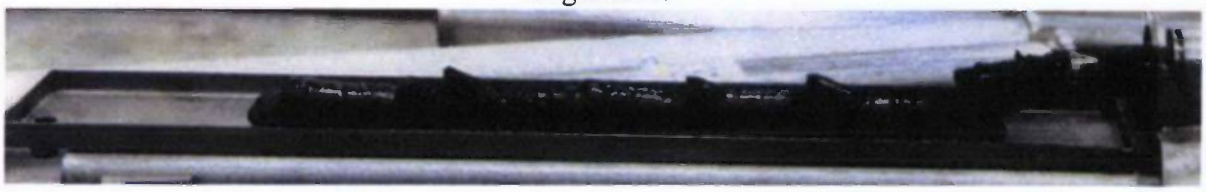
Table 3.1. Shear Strength Estimates for B- and T-200 Series Tank Core Extrusion Segments Shown in Figure 3.2

Typical Visual Descriptors	Figure 3.2 Waste Examples Tank–Core:Segment #	Estimated Shear Strength Range (Pa)
Weak; wet; slumping	T-204–188:3	30–100
Moderate strength; ductile; continuous sigmoidal extrusion	B-204–114:4	200–700
Moderate strength; more brittle; less ductile; fractured subsegments of short to medium length	B-204–114:10	700–2000
Strong; brittle; longer fractured subsegments	T-201–192:7	2000–4000

Segment #3 (toward top of waste)

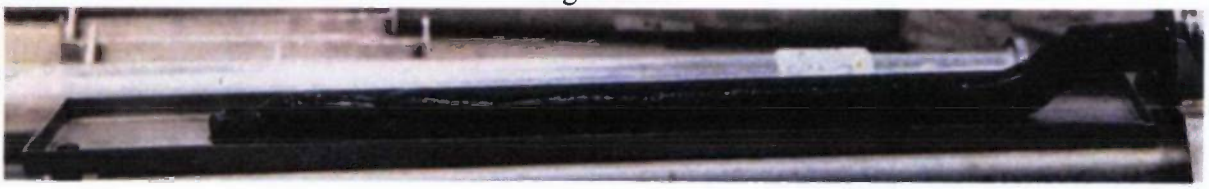
Segment #4

Segment #5



Segment #6

Segment #7



Segment #8

Figure 3.1. Photos of Core Extrusion Segments for Tank B-203, Core 122 from near the top of the waste (segment #3) to the bottom of the tank (segment #14)

Segment #9

Segment #10

Segment #11

Segment #12

Segment #13

Segment #14 (lower in tank)

Figure 3.1 (contd)

Tank T-204, Core 188, Segment #3

B-204, Core 114, Segment #4

B-204, Core 114, Segment #10

T-201, Core 192, Segment #7

Figure 3.2. Photos of Core Extrusion Segments for B- and T-200 Series Tanks from Weakest (upper photo) to Strongest (lower photo).

Of the >50 waste segments observed for 10 core extrusions from the seven TRU SSTs with photos available, the majority of waste fits in the strength range categorized by Tank B-204 core 114 segments #4 and 10 in Figure 3.2 (200–2000 Pa). For example, the bulk of the waste in Tank B-203 core 112 below segment #4 (Figure 3.1) appears to fit this categorization.[(a)] However, relatively weak wastes (30–100 Pa), like the sludge portion of segment #3 (B-203, Figure 3.1 and T-204, Figure 3.2), and relatively strong wastes (2000–4000 Pa), like segment #7 (T-201, Figure 3.2), also represent an appreciable fraction of the waste in the tanks. A few segments from all the SST photos reviewed indicate that some waste, typically nearer the bottom of the tank, may be even stronger (4000–6000 Pa estimated) than that depicted by the lower photo of Figure 3.2. The estimated waste strength as a function of location in the SST TRU waste tanks is examined in greater detail in Section 3.2.

Sludge strength variations in these tanks may be related to differences in moisture content, possibly resulting from settling and compaction. Water fractions and bulk density as a function of vertical location within the tanks are presented in Sections 5.1 and 5.2, and the relationship of estimated waste strength to these properties is explored further in Section 5.3.

3.2 Shear Strength Estimates from Extrusion Length and Slump

As presented in Section 3.1, a methodology developed by Gauglitz and Aikin (1997) allows one to estimate the rheological properties of waste sediment (solid, liquid, and gas matrix) based on a visual comparison of horizontal extrusion behavior for simulants with known yield stress in shear (or "shear strength," as it is commonly called in Hanford literature) to that of Hanford waste. Here, a related core extrusion shear strength estimation technique based strictly on extrusion length was developed from the simulant extrusion results presented in Gauglitz and Aikin. This extrusion length methodology, as well as the "slump" method, are developed and evaluated in the appendix and summarized below. The shear strength results from the application of these methodologies to the TRU SSTs are also presented.

Gauglitz and Aikin (1997) horizontally extruded bentonite/water and kaolin/Ludox/water simulants of known shear strength and reported the length at which the extrusion exhibited "failure." With these data we have the ability to correlate the shear strength of the material directly with the functional form of maximum tensile stress in a round cantilever beam:

$$\tau_y = K\frac{\rho g L^2}{d} \tag{3.1}$$

(a) The categorization of the B-203 core 112 waste is based on the photographs in Figure 3.1 and not on an assessment of initial extrusion lengths from videotape as was completed for the waste segments shown in Figure 3.2. A videotape of the B-203 core 112 extrusion was not available; the strength estimates derived from other B-203 core extrusion videos are presented in Section 3.2.

where L and d are the beam failure length and diameter, respectively, ρ is the material density, and g is the acceleration of gravity. The proportionality coefficient K of Eq. (3.1) provides a means to compute the shear strength of a material given its density and the plastic failure length of a horizontal extrusion. Plastic failure occurs in the relatively ductile bentonite simulant as necking, while the brittle kaolin/Ludox simulant fractures.

As discussed in the appendix, the proportionality coefficient is likely a function of the material microstructure. The shear strength of a material with a microstructure similar to that of the bentonite simulant is expected to be 0.89 times $\rho g L^2/d$. For materials with a microstructure similar to that of the kaolin/Ludox simulant, the shear strength is estimated to be 1.45 times $\rho g L^2/d$. As reported in Gauglitz and Aikin, the two simulants were chosen to reflect the wide variety of mechanical behaviors typical of wastes from the Hanford tanks. If we assume that the simulants "bound" the mechanical behavior of Hanford waste, we can expect that the shear strength of the waste will be between 0.89 and 1.45 times $\rho g L^2/d$. These proportionality coefficients are subsequently referred to as the "extrusion length bounds." By back-applying these results, indiscriminate of material type, to the simulant extrusion experiments, we determined that the best fit to the data is achieved with a proportionality coefficient of 1.15. This proportionality coefficient is referred to as the "extrusion length best fit."

In the event that the extruded material is so weak that it "pours" out of the sampler instead of extending out, the yield stress in shear of the material may be determined by the "slump" method, or the amount of deformation the material undergoes (Pashias et al. 1996). This methodology is detailed in the appendix.

The extrusion length and slump methodologies were applied to TRU waste Tanks B-203, B-204, T-110, and the T-200 series tanks. Horizontal core extrusion videos for these tanks were evaluated for failure length determined by the point at which failure was judged to occur. Slump measurements were also recorded where applicable.

Data taken from the upper half of a core segment are ascribed to a level in the tank corresponding to 0.75 of the segment length plus the lowest tank elevation of the segment. Similarly, data taken from the lower half of a core segment are assigned the segment elevation plus 0.25 of the segment length. Multiple measurements were available for individual segment halves in some instances, and each measurement is reported for that elevation. Waste density values, which are discussed further in Section 5.2, were taken from TWINS.[(a)]

The shear strength estimates for core 115 from B-203 are presented in Figure 3.3. Shear strengths ranging from less than 100 Pa near the top of the sediment to approximately 2,000 Pa were estimated with the extrusion length and slump techniques. An upper-bound shear strength estimate of ~2,500 Pa was determined for one B-203 core segment. These values appear

(a) Tank Waste Information System database. http://twins.pnl.gov/twins3/twins.htm.

reasonable compared with other available shear strength data for Hanford wastes (Gauglitz and Aikin 1997; Hedengren et al. 2000; TWINS). As expected due to lithostatic loading, the shear strength in B-203 tends to increase with depth, although some exceptions are noted.

Shear strength estimates for Tanks B-204 (cores 112 and 114), T-110 (core 180), T-201 (core 192), T-202 (core 191), T-203 (core 190), and T-204 (core 188) are shown in Figures 3.4 through 3.10, respectively. Results again range from less than 100 Pa to approximately 2,000 Pa for the bulk of the waste, and the shear strength increases with depth. As shown in Figure 3.7, the shear strengths of two segments of T-201 were estimated to exceed 2,000 Pa. One of these, photographed in the lower portion of Figure 3.2, had an upper-bound shear strength of ~3,500 Pa.

In B-204, an appreciable amount of data exists for two cores. Reasonable agreement in shear strength for the cores (approximately ¾ out from tank center toward the tank wall, 180° opposed), which represent unique radial locations in the tank, is shown in Figures 3.4 and 3.5.

Because of the numerous variables involved, it is not yet possible to use the core extrusion methodology to estimate strength values with a high degree of certainty. For example, the stronger waste segments of Figures 3.1 and 3.2 appear less ductile than the bentonite clay simulants and less brittle than the kaolin/Ludox simulants used as guides. Applying the extrusion technique to other simulants that more closely match the behavior of actual waste over the entire shear strength range could improve strength evaluations using the technique described here.

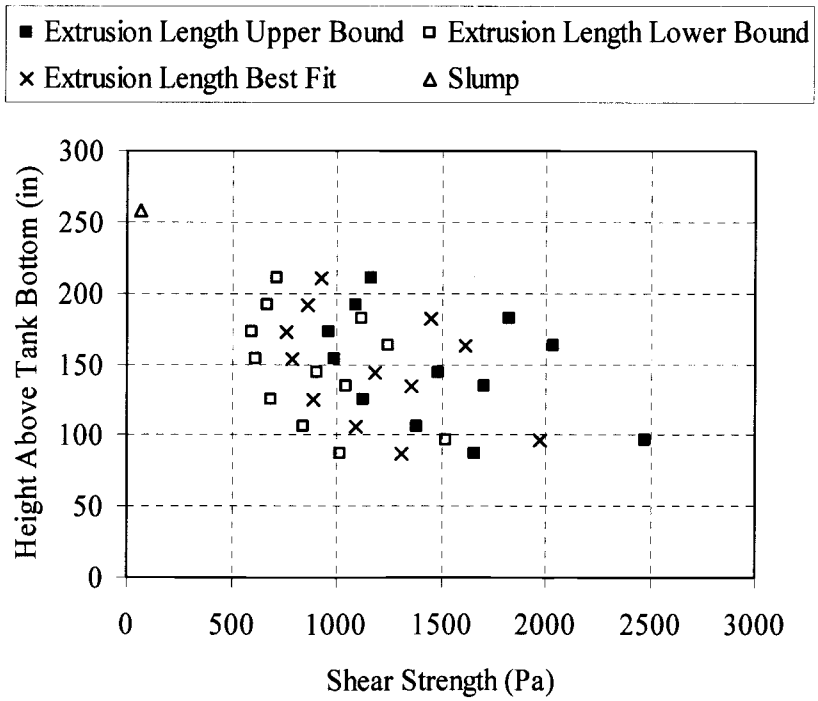


Figure 3.3. Shear Strength as a Function of Height in B-203, Core 115

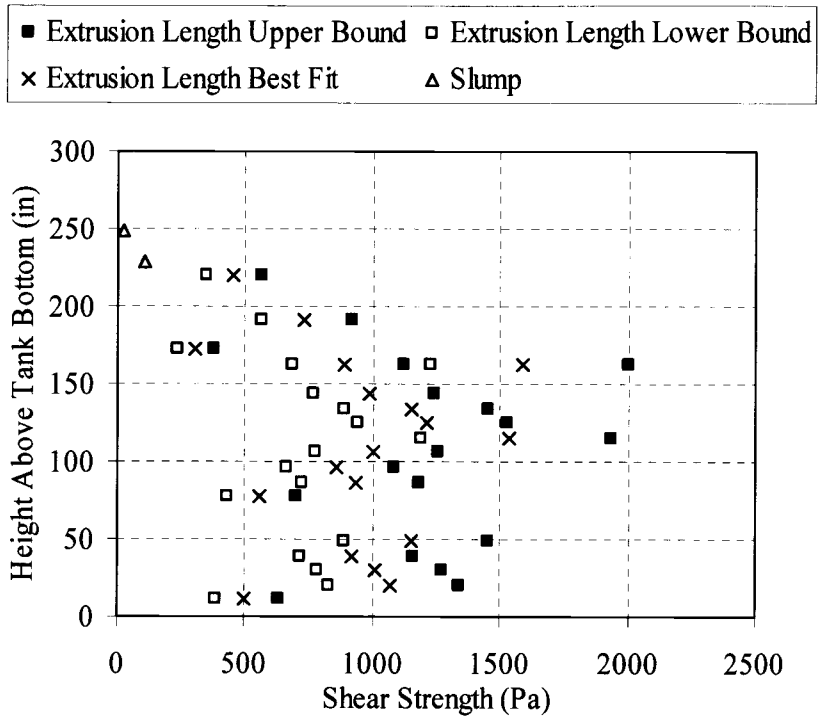


Figure 3.4. Shear Strength as a Function of Height in B-204, Core 112

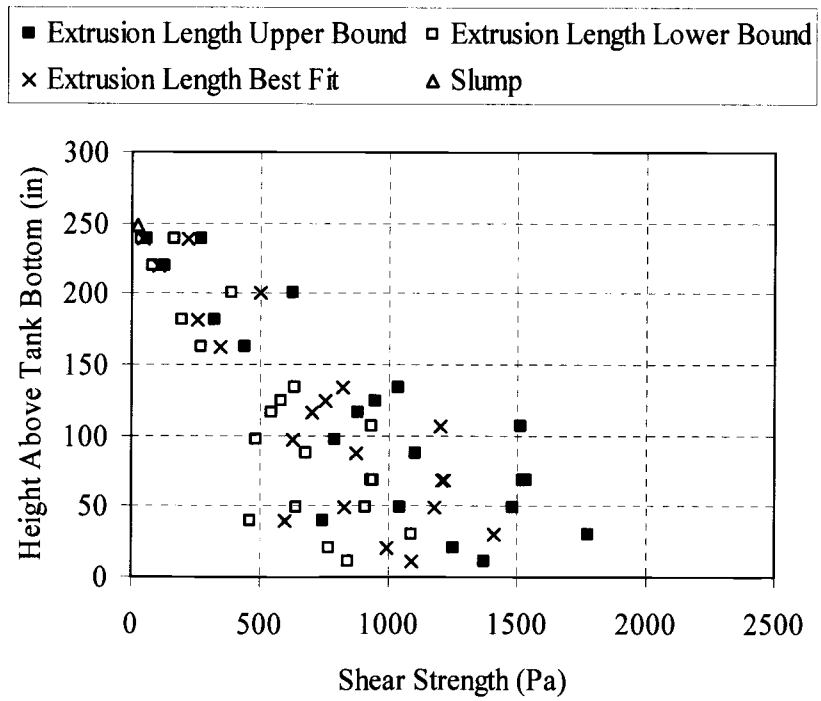


Figure 3.5. Shear Strength as a Function of Height in B-204, Core 114

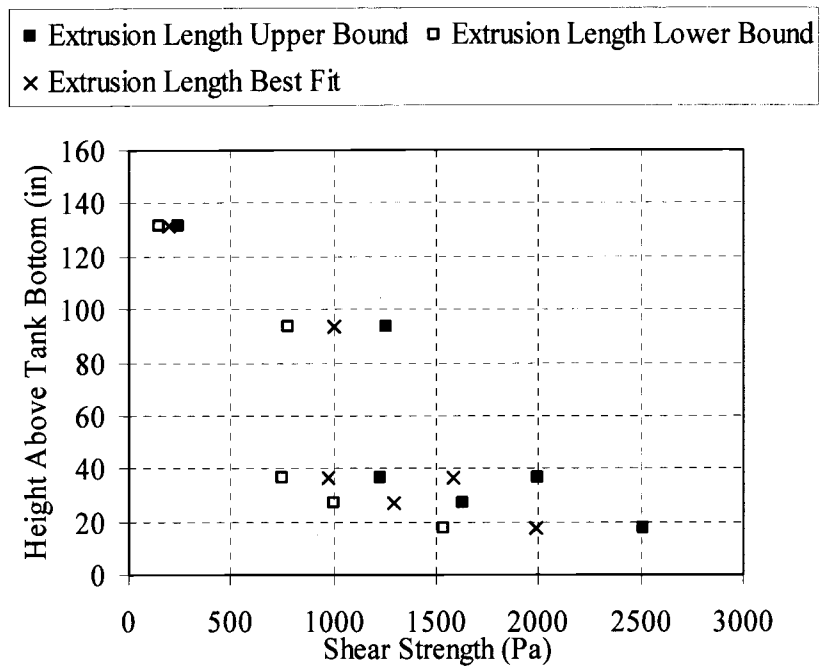


Figure 3.6. Shear Strength as a Function of Height in T-110, Core 180

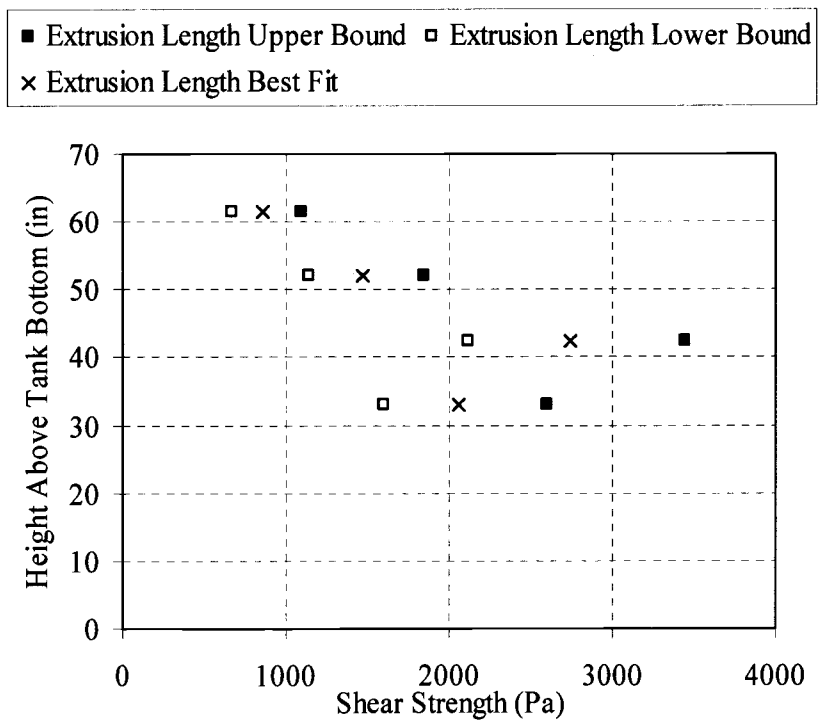


Figure 3.7. Shear Strength as a Function of Height in T-201, Core 192

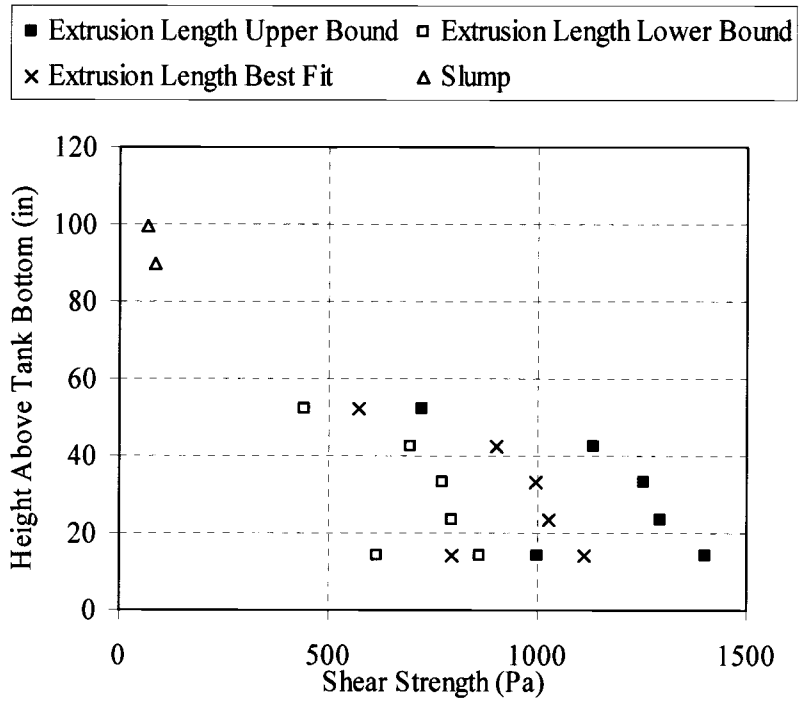


Figure 3.8. Shear Strength as a Function of Height in T-202, Core 191

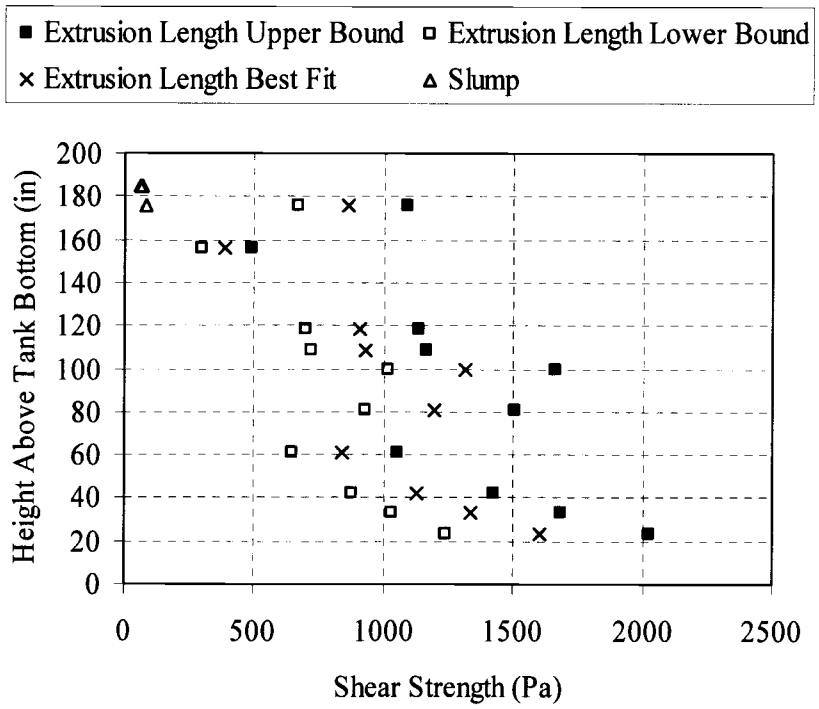


Figure 3.9. Shear Strength as a Function of Height in T-203, Core 190

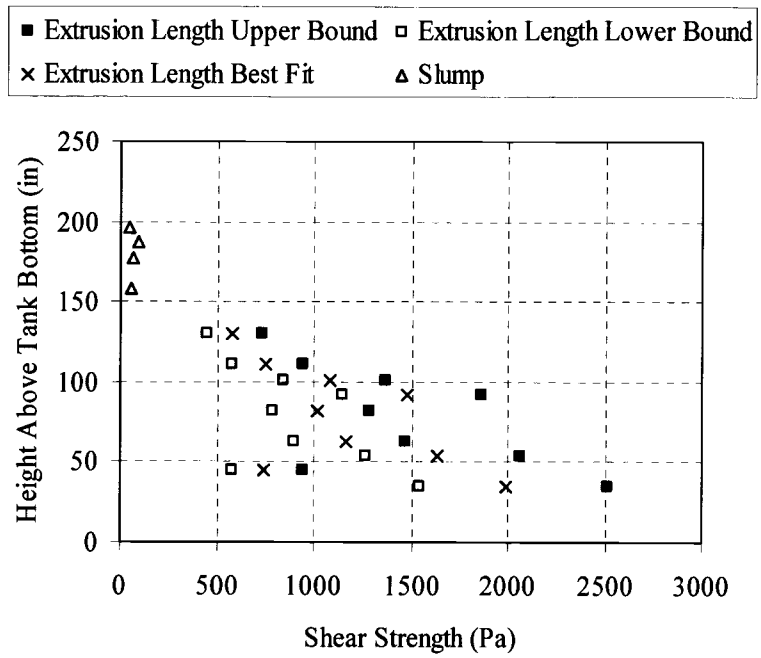


Figure 3.10. Shear Strength as a Function of Height in T-204, Core 188

3.3 Core Sampling Load Data

To obtain core samples, a sampler is pushed or rotated into the waste. The push is provided by the mass of the sampling apparatus and hydraulics on the sampler truck. It was hypothesized that this downward force could be evaluated to determine the shear strength of the waste. Rassat et al. (2000) used a similar methodology to evaluate the yield strength of the SY-101 crust layer from load data of a mechanical mitigation arm and water lance.

Investigation of the load data for the core sampling apparatus was not encouraging. Comments on the data ranged from "...down forces are independent of material properties..."[(a)] to "...don't use (load values) quantitatively...data is affected by internal friction of the sampling apparatus...."[(b)] The load data gathered from TWINS for DST and TRU tank wastes illustrate these comments. No load data trends with depth were apparent. The load data for supernatant liquid samples are counter-intuitive; higher loads are observed in the liquid than in the sediment below. Further, using just the sampler mass and the methodology in Rassat et al. (2000), the waste yield strength required to support the mass of the sampler is one to two orders of magnitude (~30,000 to 60,000 Pa) greater than other measured strengths. Based on these expectations and observations, the load data from the core sampling apparatus were not evaluated to estimate waste strength.

(a) Personal Communication from AM Templeton (CH2M HILL) to BE Wells (PNNL) on January 14, 2003.
(b) Personal Communication from J Douglas (CH2M HILL) to BE Wells on January 14, 2003.

4.0 Waste Rheology, Solids-Settling, and Particle Characterization

This section summarizes the limited amount of reported strength, viscosity, particle-characterization, and solids-settling data available for the 10 SST TRU wastes. The shear strength measurements tabulated in this section were made with viscometers, in contrast to those determined by the core extrusion observation method discussed in Section 3. The results given by the two methods are compared in this section to the limited extent possible.

4.1 Rheology

This section contains the rheological measurements that have been made on diluted waste samples from three of the TRU tanks, B-201, B-202, and T-111. Shear strength, but not viscosity, has also been measured for undiluted, unhomogenized samples. The diluted samples consistently showed pseudoplastic rheological behavior in which viscosity decreases as shear rate increases.

Tables 4.1 and 4.2 summarize rheology data for waste samples from Tanks B-201 and B-202 that were reported by Shaver (1993a, b). Several shear strength results measured with a viscometer and shear vane are included in the tables. The ranges of shear strengths were 1220 to 1410 Pa for three segments of B-201 waste and 200 to 750 Pa for six segments of B-202 waste.

Table 4.1. Rheological Data for Waste Samples from Tank B-201, Core 26[(a)]

Properties	Segment 2 (top of waste)	Segment 5 (mid-depth)	Segment 8 (bottommost)
As-received properties			
Shear strength (Pa) (Haake RV100 viscometer, M5 head and custom shear vane, 0.3 rpm)	1410	1310	1220
At 1:1 by vol dilution, 30°C			
Power-law consistency factor (Pa-s^n) (Haake RV100 viscometer, 0 to 500 s^{-1} shear rate range)	0.011	0.016	0.047
Power-law flow behavior index, n	0.86	0.92	0.80
Power-law yield stress (Pa)	1.7	5.6	8.1
Apparent viscosity (cP) at 10 s^{-1} shear rate	180	570	840
True viscosity (cP) at 10 s^{-1} shear rate	6.8	12	24
Apparent viscosity (cP) at 100 s^{-1} shear rate	23	67	100
True viscosity (cP) at 100 s^{-1} shear rate	5.0	10	15

(a) All of the waste in this core of B-201 was described as cohesive, based on its measured penetration resistance of less than 3 psi. Segment 2 was described qualitatively as varying from soft to crumbly, while lower segments were described as smooth-textured. Small pockets (<1 mL) of liquid appeared to be trapped in Segment 7, the sample above the bottommost sample (Shaver 1993a, Tables 1-3 and 1-4).

Table 4.2. Rheological Data for Waste Samples from Tank B-202, Core 24

Properties	Segment 2	Segment 4	Segment 6
As-received properties			
Shear strength (Pa) (Haake RV100 viscometer, M5 head and custom shear vane, 0.3 rpm)	200[(a)]	750[(a)]	670[(a)]
At 1:1 by vol dilution, 30°C			
Power-law consistency factor (Pa-s^n) (Haake RV100 viscometer, 0 to 500 s^{-1} shear rate range)	0.0024	0.0047	0.018
Power-law flow behavior index, n	0.98	0.94	0.76
Power-law yield stress (Pa)	0.9	1.0	1.1
Apparent viscosity (cP) at 10 s^{-1} shear rate	92	100	120
True viscosity (cP) at 10 s^{-1} shear rate	2.2	3.8	7.9
Apparent viscosity (cP) at 100 s^{-1} shear rate	11	14	17
True viscosity (cP) at 100 s^{-1} shear rate	2.1	3.4	4.5

(a) For comparison, the shear strength measurements made on Core 25 of B-202 were 270 Pa for Segment 3, 470 Pa for Segment 5, and 270 Pa for Segment 7 (Shaver 1993b, Tables 1 and 2).

These results are consistent with the range of strengths estimated from the core observations for several segments of B-203 and B-204 waste (Section 3.2), although segment 2 of B-201 core 26 is stronger than most of the near-top segments whose strengths are given in Section 3.

Rheology experiments were carried out on Tank T-111 waste including samples from segments 2 and 8 of core 31 (McKinley et al. 1992). Tests were also done on segment 4, but the sample had dried out prior to testing, and the results were not considered representative. Applying the same test methods used on the B-201 and B-202 samples, the shear strength of the undiluted T-111 samples was 500 ± 230 Pa.

A limited amount of rheological testing was performed on waste from Tanks B-201 and B-202 that had been diluted 1:1 (volume basis) with water. The rheological parameters were obtained by curve-fitting data for shear stress versus strain rate to a yield power law function. The form of the yield power-law function is

$$\tau = \tau_y + K\gamma^n$$

where

τ = shear stress in fluid (Pa)

τ_y = yield stress (Pa)

K = consistency factor (Pa s^n)

n = flow behavior index

γ = shear rate (s^{-1}).

The "apparent" viscosity values in Tables 4.1 and 4.2 were calculated as τ/γ and are most accurate for high shear rates. The "true" viscosity values were calculated as the derivative of τ with respect to γ. For pipe flow, the shear rate at the wall is roughly equal to 8 V/D, in which V is the average flow velocity and D the pipe diameter (Wasp et al. 1977); this relation can be used to estimate the shear rate so that the viscosity at that condition can be calculated.

In Tank B 201, whose rheological data are summarized in Table 4.1, the viscosity and the power law yield stress of the diluted waste increases from top to bottom (Shaver 1993a). At a strain rate of 100 s^{-1}, the true viscosity of the diluted waste ranges from ~0.0050 to 0.015 Pa-s (5 to 15 cP). Table 4.2 presents similar data for waste samples from Tank B-202 (Shaver 1993b). The diluted samples from B-202 are somewhat lower in power law yield stress and viscosity [~0.0021 to 0.0045 Pa-s (2.1 to 4.5 cP) at 100 s^{-1}] than those from B-201. In B-202, as in B-201, the viscosity increases from the top to the bottom of the waste.

The true viscosity of the diluted waste in B-201 and B-202, as computed from the derivatives of their respective power law functions from Tables 4.1 and 4.2, is presented in Figures 4.1 and 4.2, respectively. Note that data obtained from the typically used viscometers may be questionable for strain rates of less than 50 s^{-1} (e.g., Herting 1999). At a constant strain rate, viscosity increases with waste depth (Figures 4.1 and 4.2).

The increase in viscosity with depth is expected if we attribute decreased water content in the waste with depth, as is shown in Section 5.1, to increased undissolved solids content. For a given waste, we expect a sample with a higher undissolved solids content to also have a higher undissolved solids content after dilution than a sample having a lower initial undissolved solids content and diluted equivalently. Typically, at a fixed strain rate, the viscosity of a mixture will increase with increasing undissolved solids volume fraction. This may be seen in Einstein-type correlations, which relate slurry viscosity to the undissolved solids volume fraction and base liquid viscosity (Wasp et al. 1977). The effect of temperature on slurry viscosity is not addressed in the current discussion.

To relate the diluted waste sample viscosity results to in situ waste conditions from the viewpoint of increased undissolved solid volume fraction, examination of data for other Hanford wastes is insightful. Data from DST saltcake waste suggests that, at very low strain rates (<1 s^{-1}, data from the ball rheometer) (Stewart et al. 1996b), halving the undissolved solid volume fraction (by dilution, for example) could reduce the slurry viscosity by as much as four orders of magnitude (Onishi et al. 2003a, b). The effect of the altered base liquid viscosity by dilution is negligible. At a strain rate of approximately 100 s^{-1}, the effect of the halved solids volume fraction is reduced to approximately one to two orders of magnitude (Onishi et al. 2002, 2003a, b), and at 1,000 s^{-1}, the effect is reduced to less than 0.025 Pa-s (25 cP). Qualitatively, therefore, the argument may be made that the viscosity values of Figures 4.1 and 4.2 are likely lower than the waste at in situ conditions for a given strain rate.

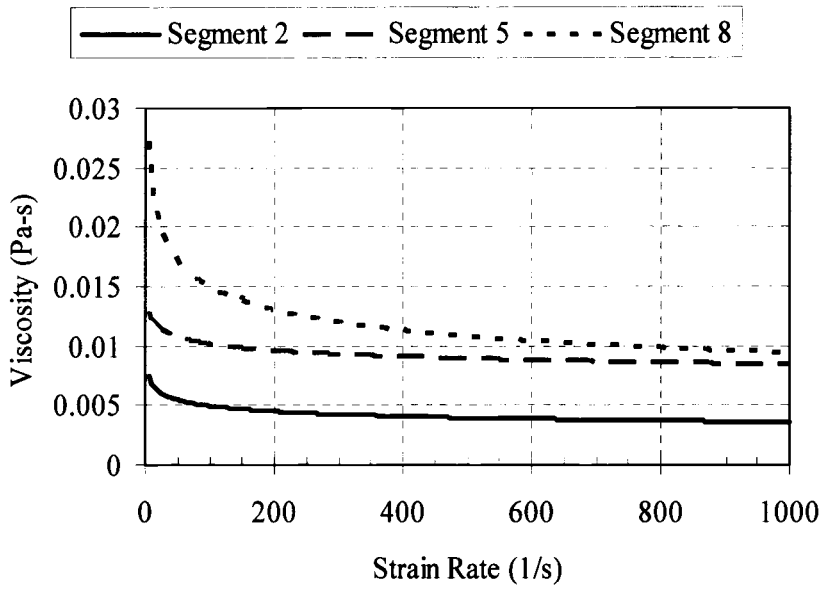


Figure 4.1. Viscosity as a Function of Strain Rate for 1:1 Diluted B-201 Samples (core 26, parameters from Table 4.1)

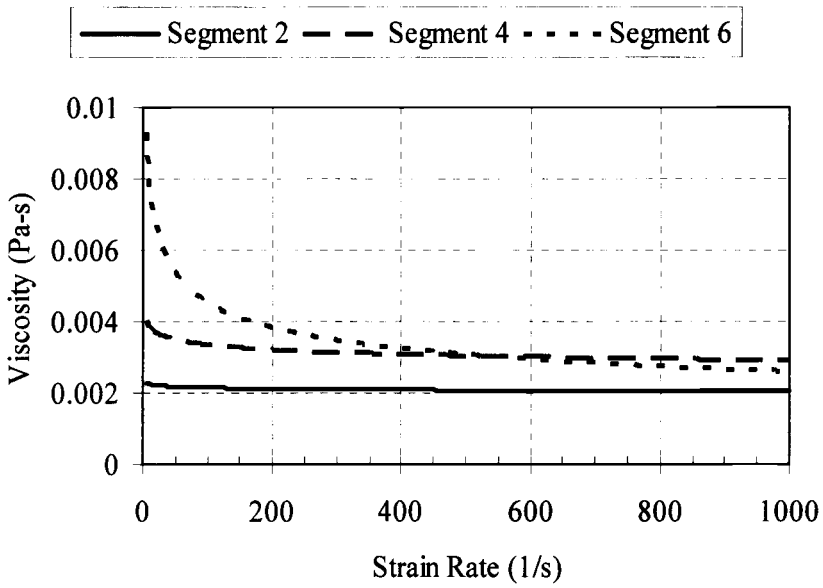


Figure 4.2. Viscosity as a Function of Strain Rate for 1:1 Diluted B-202 Samples (core 24, parameters from Table 4.2)

The viscosity of diluted and undiluted samples of T-111 was also investigated (McKinley et al. 1992). Rheograms (shear stress versus strain rate) showed that the undiluted T-111 samples behaved like greases and lubricants: the viscometer cone tended to slip over the sample, with the slippage increasing with shear rate. As a result, the rheological behavior of these samples could not be described in power-law form. The viscosity of the T-111 samples was also measured for 1:1 and 3:1 sample dilutions with water. The diluted samples evidenced pseudoplastic behavior. Because the viscosities were near the detection limit of the system (2 cP), the data could not be correlated and power-law constants are not available.

4.2 Solids-Settling

Design issues related to the development of a TRU waste dewatering process can be partially addressed by solids-settling data for the wastes. This section discusses the available settling data. Simple solids-settling tests were conducted for undiluted and diluted samples from B-201, B-202, and T-111 (Shaver 1993a, b; McKinley 1992). The results of the tests are summarized in Tables 4.3, 4.4, and 4.5. The tests showed that at 1 G (gravitational force on Earth), under the small hydrostatic head available in a centrifuge tube, essentially no settling occurred in the undiluted samples. Centrifugation of the undiluted samples at more than 1000 G produced 2 to 12 vol% free liquid in B-201 samples and 30 to 40 vol% free liquid in B-202 samples. This may represent an upper bound for the amount of waste settling and dewatering that could be produced by vibration or other disturbances.

Both as-received and water-diluted samples showed potential for settling under their own weight and for centrifugal dewatering. The data in Shaver (1993a, b) and McKinley et al. (1992) suggest that the rate of gravitational settling for both waste types decreased significantly after two days, but centrifugation clearly indicates that settling was not complete. In two days, gravity settling produced free liquid of 8 to 19 vol% in 1:1 (by volume) diluted B-201 samples, whereas centrifugal settling increased the free liquid to 42 to 58 vol%. Likewise, the free liquid in 3:1 diluted B-201 samples was 43 to 63 vol% after 30 hr at 1 G and 76 to 84 vol% when centrifuged. Dissolution of salt solids may contribute to the reduced total mass of solids in diluted samples.

Studies of other TRU wastes that have high metals content, though considerably different in composition than those shown in Table 2.1, have suggested that the volume of long-settled sludge would be half that of sludge settled for only a few days (Swanson 1991). The applicability of these studies to the particular TRU wastes under consideration in this report is unknown, but this observation seems roughly consistent with the decrease in volume between >1-day gravity settled solids volumes and the centrifuged solids volumes shown in Table 4.3 for 1:1 and 3:1 diluted B-201 samples.

Table 4.3. Density and Settling Data for Waste Samples from Tank B-201, Core 26[a]

Properties	Segment 2 (top of waste)	Segment 5 (mid-depth)	Segment 8 (bottommost)
As-received properties			
Centrifuged bulk solids density (g/cc) (centrifuged for 1 hr at >1000 G)	1.66	1.52	1.37
Centrifuged liquid density (g/cc)	1.19	1.19	1.05
Settled solids (vol%) (settling at 1 G in centrifuge tube)	100	100	100
Centrifuged solids (vol%)	98	98	88
Centrifuged solids (wt%)	98	98	90
At 1:1 dilution by volume, 30°C			
Pre-centrifugation bulk density (g/cc)	1.33	1.17	1.13
Centrifuged bulk solids density (g/cc) (centrifuged for 1 hr at >1000 G	1.59	1.40	1.33
Centrifuged liquid density (g/cc)	1.01	1.00	0.99
Settled solids (vol%) (after 8 hr at 1 G in centrifuge tube)	90	88	94
Settled solids (vol%) (after 48 hr at 1 G in centrifuge tube)	83	81	92
Centrifuged solids (vol%)	58	43	42
Centrifuged solids (wt%)	69	52	49
At 3:1 dilution by volume, 30°C			
Pre-centrifugation bulk density (g/cc)	1.10	1.05	1.05
Centrifuged bulk solids density (g/cc) (centrifuged for 1 hr at >1000 G)	1.48	1.36	1.24
Centrifuged liquid density (g/cc)	0.99	1.00	0.99
Settled solids (vol%) (after 8 hr at 1 G in centrifuge tube)	48	40	60
Settled solids (vol%) (after 30 hr at 1 G in centrifuge tube)	42	37	57
Centrifuged solids (vol%)	24	16	21
Centrifuged solids (wt%)	32	21	25

(a) Data taken from Tables 1-3 to 1-5 and Figures 1-17 to 1-19 of Shaver (1993a).

Table 4.4. Density and Settling Data for Waste Samples from Tank B-202, Core 24[(a)]

	Segment 2	Segment 4	Segment 6
As-received properties			
Pre-centrifugation bulk density (g/cc)	1.23	1.20	1.21
Centrifuged bulk solids density (g/cc) (centrifuged for 1 hr at >1000 G)	1.32	1.29	1.29
Centrifuged liquid density (g/cc)	1.06	1.05	1.03
Settled solids (vol%) (settling at 1 G in centrifuge tube)	100	100	100
Dissolved & undissolved solids (vol%)	36	28	27
Undissolved solids (wt%)	33	24	24
Centrifuged solids (vol%)	67	61	69
Centrifuged solids (wt%)	72	65	73
At 1:1 dilution by volume, 30°C			
Settled solids (vol%) (after 8 hr at 1 G in centrifuge tube)	84	87	92
Settled solids (vol%) (after 55 hr at 1 G in centrifuge tube)	77	80	87
At 3:1 dilution by volume, 30°C			
Settled solids (vol%) (after 8 hr at 1 G in centrifuge tube)	43	45	54
Settled solids (vol%) (after 55 hr at 1 G in centrifuge tube)	39	40	49

(a) Data taken from Tables 1 and 2 and Figures 1 to 3 of Shaver 1993b.

Table 4.5. Density and Settling Data for Waste Samples from Tank T-111, Core 31[(a)]

As-received properties	Segment 2	Segment 8
Pre-centrifugation bulk density (g/cc)	1.19	1.28
Centrifuged bulk solids density (g/cc) (centrifuged for 1 hr at >1000 G)	1.22	1.34
Centrifuged liquid density (g/cc)	1.07	1.10
Settled solids (vol%) (settling at 1 G in centrifuge tube)	100	100
Dissolved & undissolved solids (wt%)	22.4	29.3
Undissolved solids (wt%)	19.0	25.4
Centrifuged solids (vol%)	65.8	71.9
Centrifuged solids (wt%)	67.3	75.9
At 1:1 dilution by volume, 30°C		
Settled solids (vol%) (after 8 hr at 1 G in centrifuge tube)	92	96
Settled solids (vol%) (after 60 hr at 1 G in centrifuge tube)	87	80
At 3:1 dilution by volume, 30°C		
Settled solids (vol%) (after 8 hr at 1 G in centrifuge tube)	65	58
Settled solids (vol%) (after 52 hr at 1 G in centrifuge tube)	52	40

(a) Data taken from Field (1997, Tables B2-10 - 11) and McKinley et al. (1992) Figures 1 - 4.

4.3 Particle Characterization

Particle size in the waste and the abrasiveness of the particles may have significance in the development of TRU waste handling and packaging equipment and the physical simulants needed to test the process. The available data on particle properties of the SST TRU wastes from Tanks B-201, B-202, and T-111 are presented in this section.

The particle size distribution in the waste from B-201 was measured in glycerol dispersions made from unhomogenized subsamples of each segment using a laser-scan transit-time technique. The instrument was a Brinkman Model 2010 particle size analyzer (Shaver 1993a). A summary of particle size distribution[(a)] is given in Table 4.6. Particle size distributions were measured by similar methods in T-111 core 31, and the results are summarized in Table 4.7.

Figures 4.3 through 4.10[(b)] show the particle size distributions in each segment of core 27 in volume percent. The segments are given in order of decreasing elevation; segment 1 (27:1) is

Table 4.6. Particle Size Distribution Data for Waste Samples from Tank B-201

Core:Segment	Particle Size Distribution, by Volume[(a)]		Particle Size Distribution, by Number[(a)]	
	Mean (μm)	Median (μm)	Mean (μm)	Median (μm)
26:1	31.3	13.8	1.22	0.90
26:2	22.4	13.8	1.32	0.93
26:3	29.2	27.9	1.21	0.91
26:4	10.7	5.28	0.99	0.82
26:5	28.4	26.3	1.26	0.92
26:6	38.7	43.1	1.14	0.88
26:7	20.4	12.2	1.23	0.93
26:8	6.83	4.83	0.98	0.81
27:1	26.4	20.0	1.13	0.88
27:2	65.5	46.6	1.31	0.91
27:3	30.5	21.6	1.48	0.92
27:4	18.0	12.1	1.07	0.84
27:5	9.42	6.46	1.16	0.87
27:6	41.8	37.5	1.56	1.03
27:7	18.6	17.4	1.24	0.93
27:8	23.2	17.6	1.10	0.86

(a) Tank Waste Information Network System (TWINS), http://twins.pnl.gov:8001/data/data.asp.

(a) Tank Waste Information Network System (TWINS), http://twins.pnl.gov:8001/data/data.asp.
(b) McKinley SG, LR Greenwood, EW Hoppe, RT Steele, JM Tingey, and MW Urie. May 7, 1993. *Core 27 Data Report, Tank B-201* Rev. 0, Pacific Northwest Laboratory, Richland, WA.

Table 4.7. Particle Size Distribution Data for Waste Samples from Tank T-111

Core:Segment	Particle Size Distribution, by Volume[a]		Particle Size Distribution, by Number[a]	
	Mean ± Std. Dev. (μm)	Median (μm)	Mean ± Std. Dev. (μm)	Median (μm)
31:1	28.6 ± 35.9	5.8	1.23 ± 0.89	0.94
31:2	14.9 ± 20.8	4.8	1.13 ± 0.80	0.88
31:3	65.0 ± 46.2	58.7	1.17 ± 1.00	0.91
31:4	24.9 ± 34.2	5.6	0.93 ± 0.60	0.80
31:5	37.9 ± 47.9	12.3	0.95 ± 0.63	0.81
31:7	8.0 ± 11.9	4.0	0.97 ± 0.60	0.83
31:8	24.7 ± 28.2	10.0	1.02 ± 0.85	0.82
31:9	59.7 ± 49.0	59.0	1.02 ± 0.83	0.83

(a) Field (1997), Tables B2-8 and B2-9.

the top segment and segment 8 (27:8) the bottom. Tank waste segments are 19 inches apart. The reason for the irregular particle size variation from segment to segment is unknown. Particle size distributions for T-111 waste samples, though not identical, exhibit a range of behavior similar to that of B-201 in Figures 4.3 to 4.10.

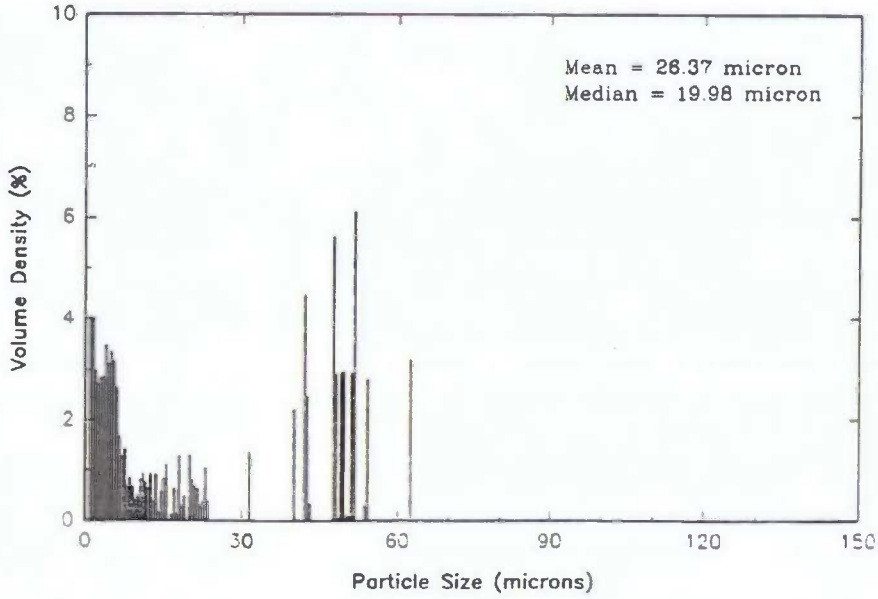


Figure 4.3. Particle Size Distribution in Core 27, Segment 1 of B-201

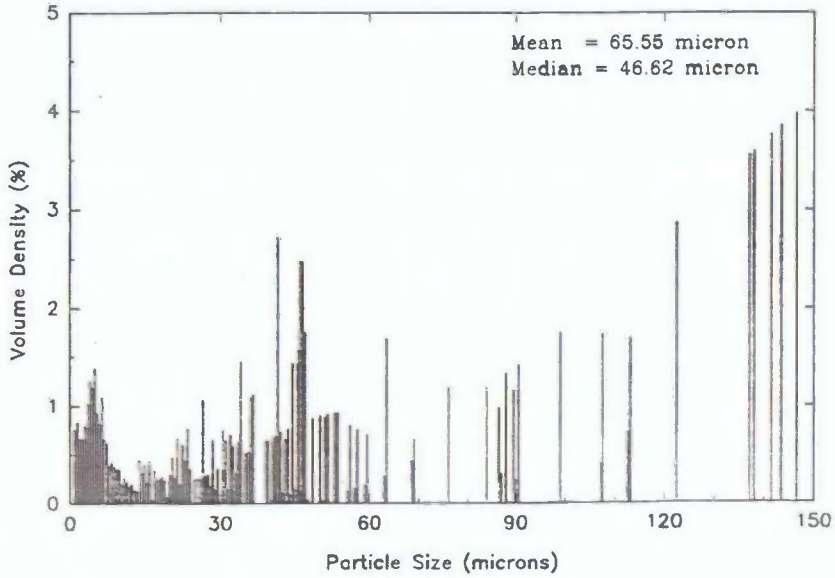


Figure 4.4. Particle Size Distribution in Core 27, Segment 2 of B-201

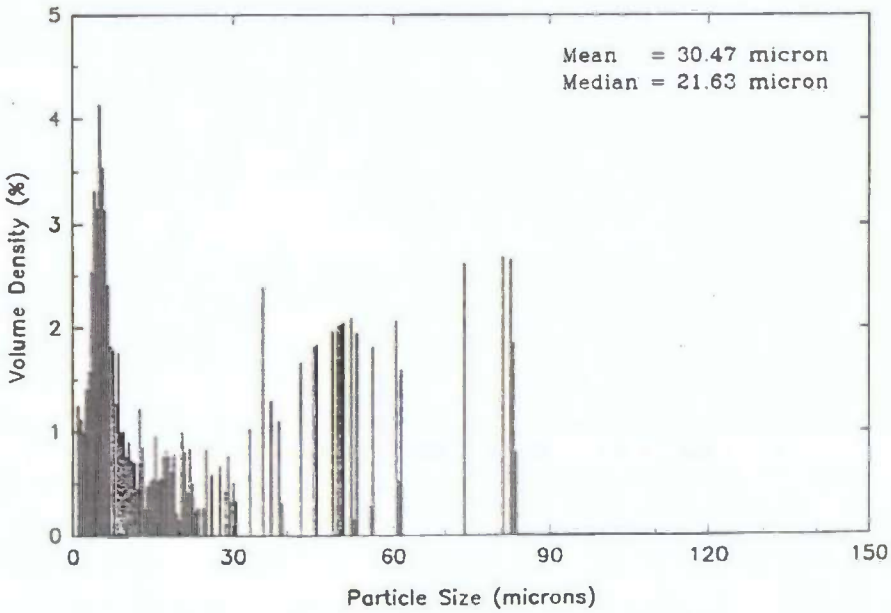


Figure 4.5. Particle Size Distribution in Core 27, Segment 3 of B-201

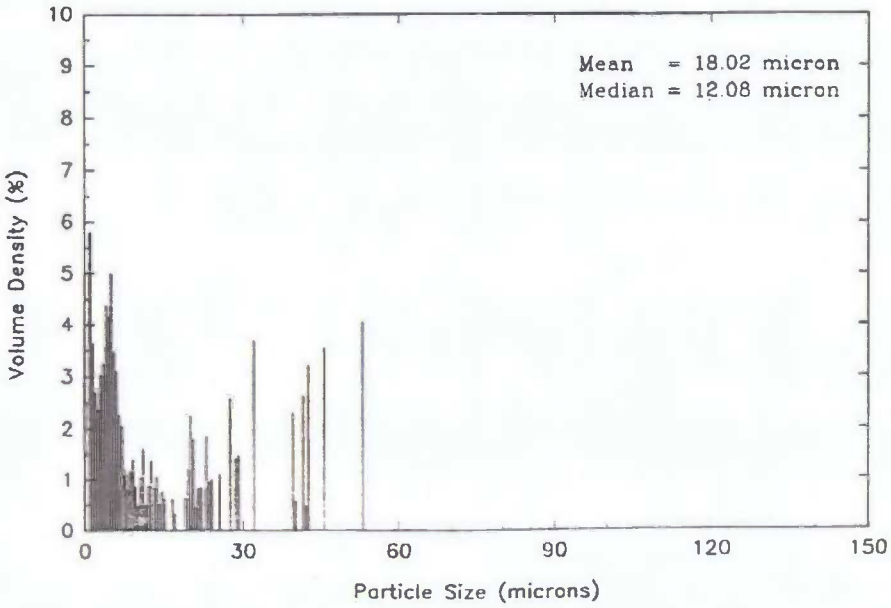


Figure 4.6. Particle Size Distribution in Core 27, Segment 4 of B-201

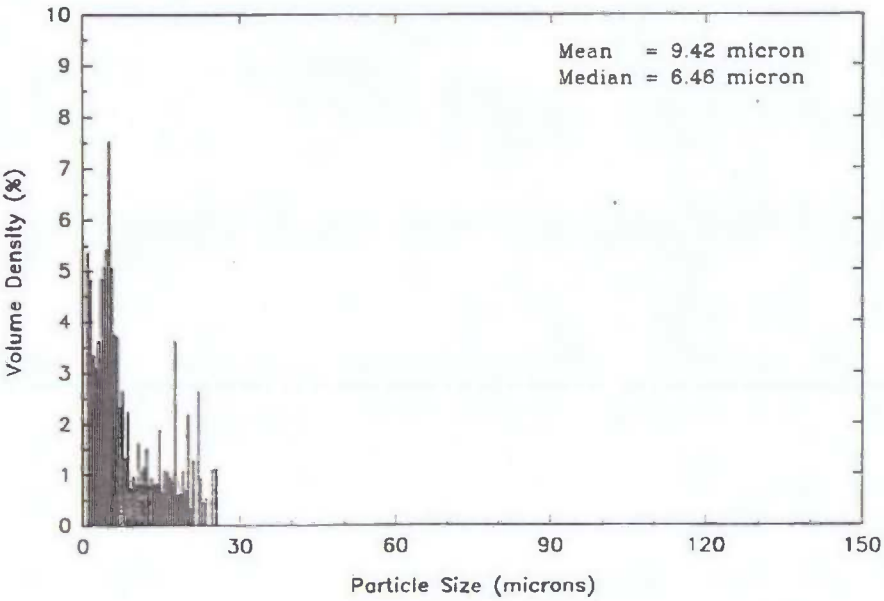


Figure 4.7. Particle Size Distribution in Core 27, Segment 5 of B-201

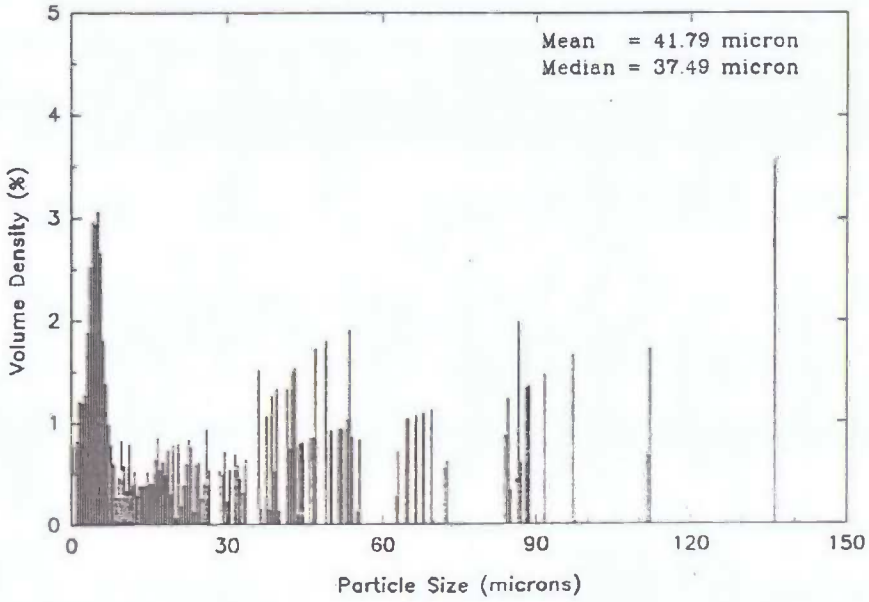


Figure 4.8. Particle Size Distribution in Core 27, Segment 6 of B-201

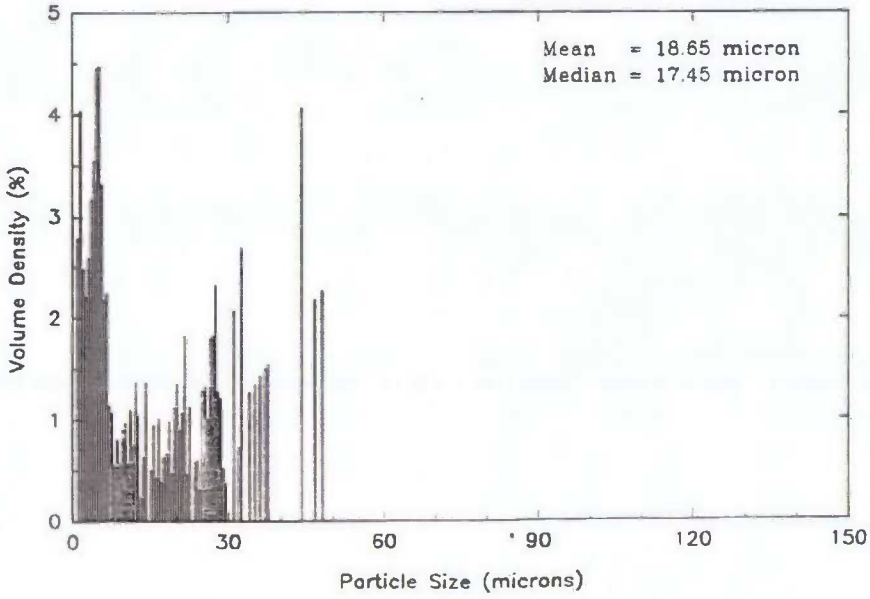


Figure 4.9. Particle Size Distribution in Core 27, Segment 7 of B-201

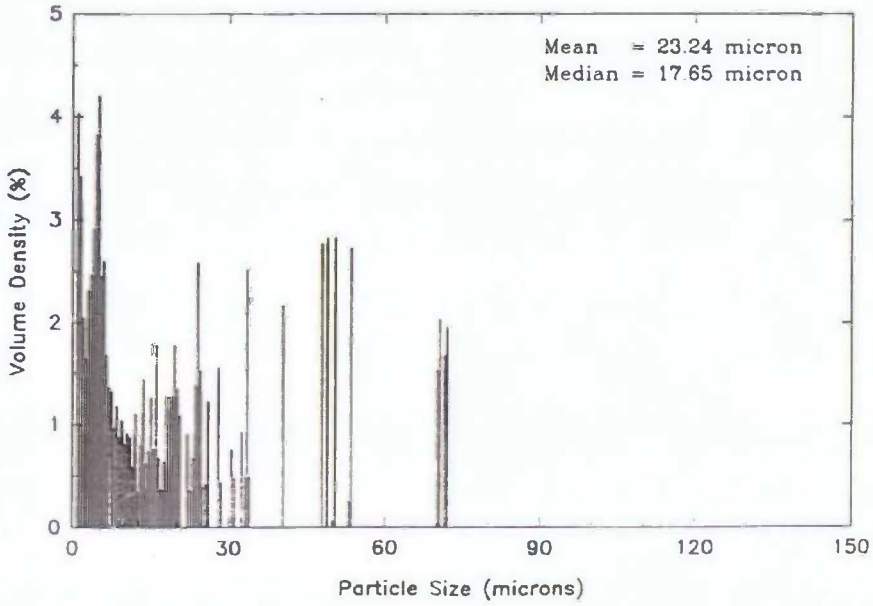


Figure 4.10. Particle Size Distribution in Core 27, Segment 8 of B-201

In one of the Tank B-201 composites from core 27, leaching studies suggested that the large particles (>6 μm) were composed mainly of compounds removed by caustic leaching. These were mostly sodium salts, probably including sodium nitrate, phosphate, and oxalate (Lumetta and Rapko 1994). This suggests a distinction in composition between larger and smaller particles.

Another property of some significance is the abrasiveness of the waste. A general discussion of waste abrasiveness can be found in Duignan (2002), and the points of interest are summarized here. Abrasiveness is linearly proportional to the Vickers Hardness of the solids in the waste. It is linearly proportional to the particle size for particle diameters less than 100 μm but does not vary with particle diameter above 100 μm. It is linearly proportional to the solids mass fraction for less than 5 wt% solids, still linearly proportional between 5 and 20 wt% but with a smaller proportionality constant, and increases only slightly for solids concentration above 20 wt%.

Information about the compounds present in the tank solid phase is obviously important to estimating or matching the waste abrasiveness. Of the tanks whose waste is being considered for dry retrieval, the waste from Tanks B-202 and T-111 has undergone phase characterization by a combination of techniques, including powder X-ray diffraction (XRD), transmission electron microscopy/electron diffraction (TEM/ED), and scanning electron microscopy/electron dispersive X-ray (SEM/EDX). Rapko and Lumetta (2000) described details of the techniques.

XRD did not identify any of the solid phases in the B-202 sample except for $NaNO_3$ (sodium nitrate). TEM, SEM, and XRD techniques were used on the T-111 waste sample. They identified Na_3PO_4, $La_4(P_2O_7)_3$, $Ca_5OH(PO_4)_3$ (a form of apatite), $BiPO_4$, $FePO_4$, amorphous $Fe(OH)_3$, Mn_2MnO_4, Fe_2MnO_4 (jacobsite), and FeOOH (goethite). The presence of other solid phases is not precluded by the fact that they were not identified, but they are less likely to be present in significant quantities. Of the phases that were identified, those with the greatest hardness were apatite (5 Mohs hardness) and goethite (5 to 5.5 Mohs). The manganese-containing phases may also be hard, and any silicates that were present but not identified could be harder than goethite.

5.0 Relationship of Water Content, Bulk Density, and Shear Strength

Evidence of waste compaction (dewatering) is shown by measurements of the water content and, to some extent, the bulk density of SST TRU waste samples. Additionally, waste shear strength variations in the tanks, as discussed in Section 3, suggest a degree of compaction. The data were examined for correlation between water content, bulk density, and shear strength; only slight indications of correlation were found.

5.1 Water Content

The bulk water content of core samples from the various TRU SSTs have been measured.[(a)] ("Bulk" indicates that both solids and interstitial liquid are included.) All the water fraction data presented in this section were obtained by thermogravimetric analysis (TGA); in some cases, especially in the T-200 tanks, this method was supplemented by differential scanning calorimetry (DSC). The TGA method determines water content by measuring the weight loss resulting from progressive heating of the sample. In most of the samples, the water loss occurred below 130°C; in this temperature range the water probably came from liquid. In some samples, a small amount of the water (15% or less) did not evaporate until temperatures between 130°C and 300°C were reached; this water probably represents chemically bound water of hydration in the solids. Thus the maximum overstatement of liquid water content that might result from including the water of hydration is about 15%.

In a few cases, the water content of samples may have been underestimated because the sample-handling procedure allowed free liquid to drain from the sample. (The liquid was thereafter handled and analyzed separately.) However, the samples of the wastes discussed in this report produced little or no free liquid, so drainage was not a concern in data analysis.

Figure 5.1 provides information on the water content profiles in the tanks, hence on the long-term waste settling. The plots in the figure show the weight percent water as a function of the depth below the waste surface for all the TRU SSTs except B-201 and B-202, for which water content data were unreliable. A decrease in the water content with depth, suggestive of waste self-compaction, can be observed in most of these tanks, with T-201 and T-203 as possible exceptions.

(a) All data not otherwise referenced were taken from the Hanford Tank Waste Information Network System (TWINS) database, http://twins.pnl.gov:8001/twins.htm.

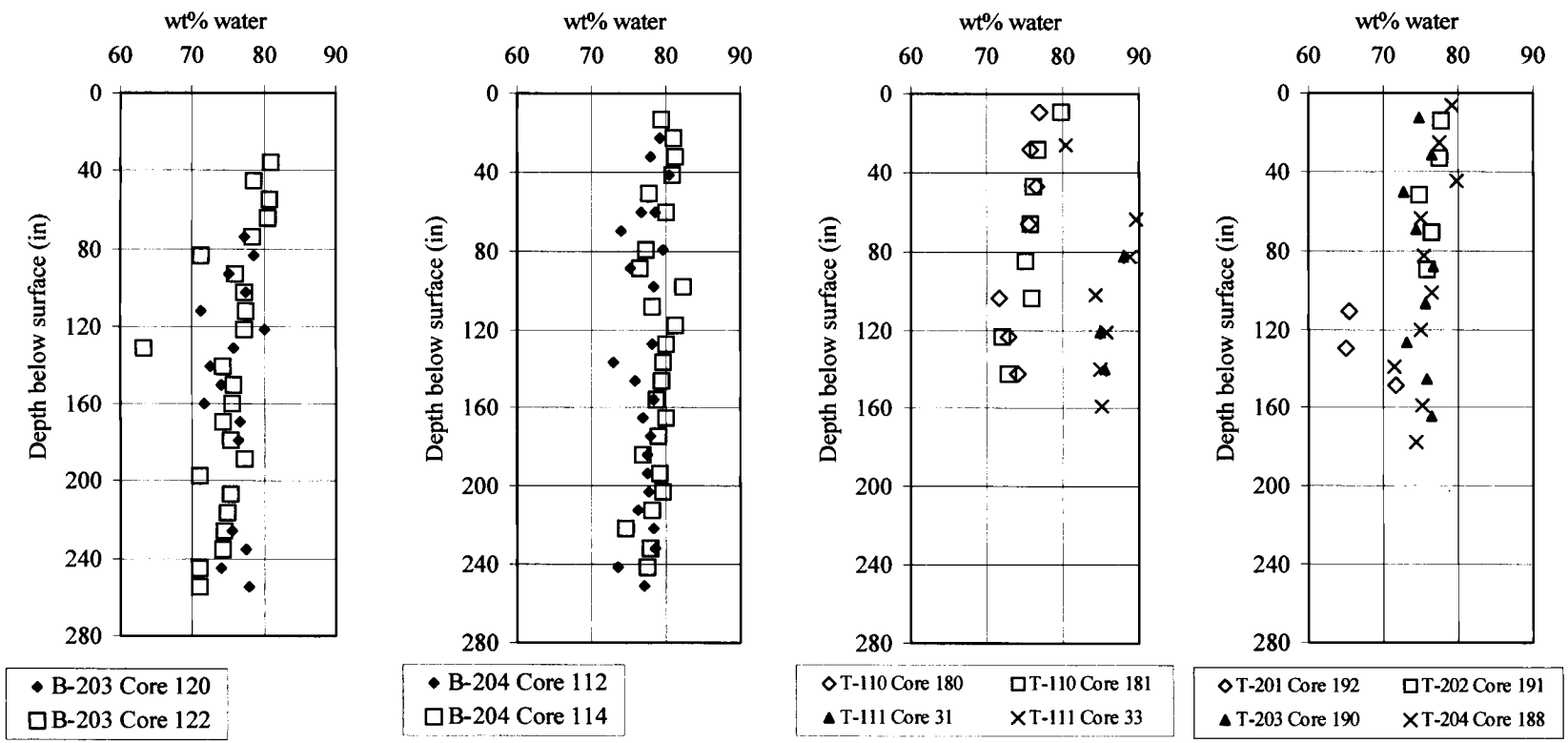


Figure 5.1. Waste Compaction, as Indicated by Variation in Water Content

5.2 Bulk Density

The bulk density of the waste depends on the waste liquid fraction and the densities of the liquid and solid phases. This section compares the bulk density with the water fraction (which is nearly equal to the liquid fraction). Figure 5.2 shows the relationship of the sample bulk densities and water fractions (the latter are expressed as weight percent water) in the two B-200 series tanks for which reliable density and water data are available. (Some density and water data for Tanks B-201 and B-202 were also found, but these did not show good repeatability.) The data points in the upper-left corner of the plot are for tank liquid (typically containing inconsequential amounts of suspended solids). There is marked scatter in the data in Figure 5.2, but it appears that the density increases more rapidly with decreasing water content in B-204 waste than in B-203 waste.

Figures 5.3 and 5.4 show the density/water relationships for the T-200 series and T-110 and T-111 tanks. Considering the three data sets together (Figures 5.2 through 5.4), most of the data for bulk solids samples show bulk densities between 1.15 and 1.35 g/mL and water content between 70 and 85 wt%. Tanks B-203, T-201, and T-204 seem to have similar density/water relations. Tanks B-204, T-203, T-110, and T-111 show a greater amount of density change per decrease in water content than does the first group. Tank T-202 shows densities that vary even though the water contents of the various samples are about the same.

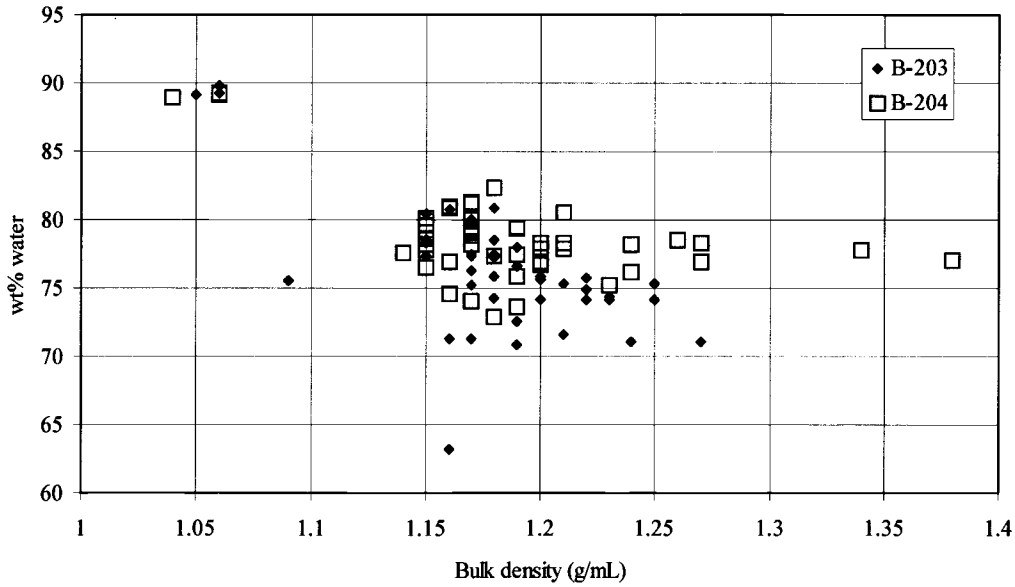


Figure 5.2. Bulk Density Versus Water Content in the B-200 Series Tanks

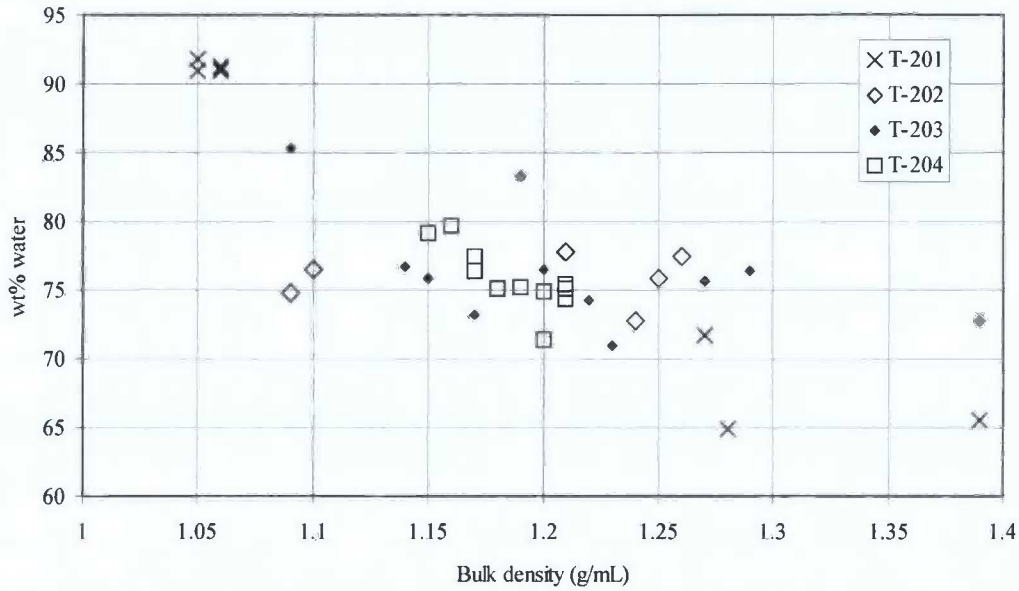


Figure 5.3. Bulk Density Versus Water Content in the T-200 Series Tanks

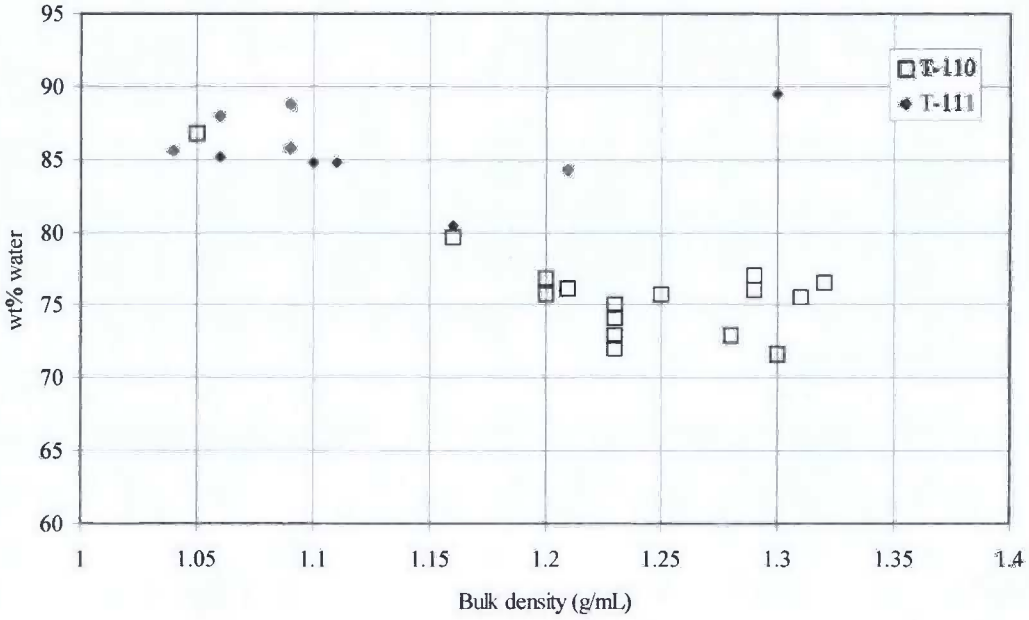


Figure 5.4. Bulk Density Versus Water Content in Tanks T-110 and T-111

5.3 Waste Shear Strength

The best-estimate shear strength data from Section 3.2 are compared with water content and bulk density data for the same tank core segments, where overlapping data exist, to determine whether a meaningful correlation exists. Figures 5.5 and 5.6 show the best-estimate waste shear strength plotted with water content and density, respectively. The shear strength shows some tendency to decrease with increasing water, but the scatter in the data overwhelms the trend. There is an even less-evident relationship between the shear strength and the density.

One potential source of error in these comparisons results from the variation in strength within a given core segment. Whereas bulk density and water content are measured on a homogenized portion of a half-core segment, the shear strength estimates were obtained only for portions of the half-core segments with measurable extrusion failure lengths. In other words, the shear strength estimates are characteristic of the waste in the vicinity but do not represent averages over the core segment.

As noted in Section 3.2, the data suggest that the waste shear strength increases with depth in the tank in much the same way that water content and bulk density vary with depth. While water content may be a factor in waste shear strength (and bulk density), many other unspecified factors could have significant roles. For example, particle size, shape, and crystal structure are likely to impact shear strength.

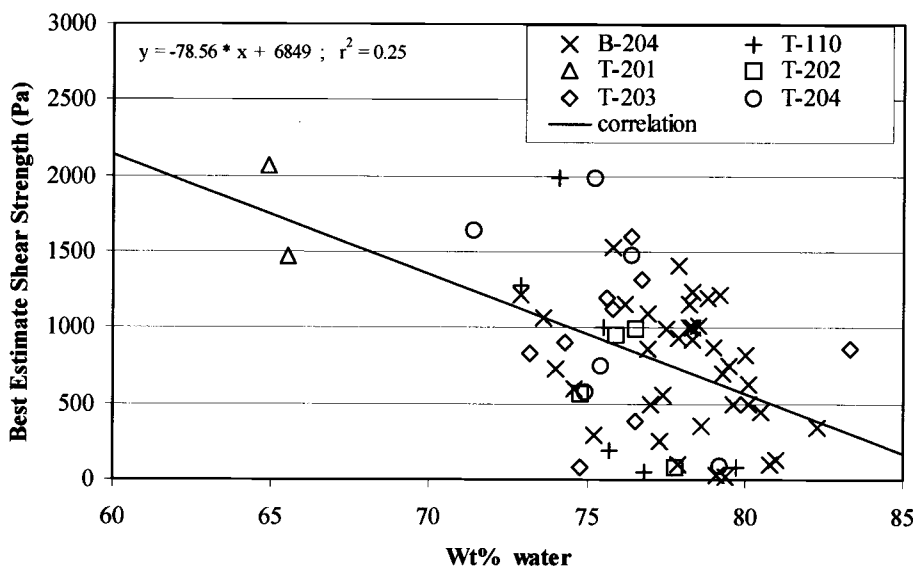


Figure 5.5. Relation of Estimated Waste Shear Strength and Water Content

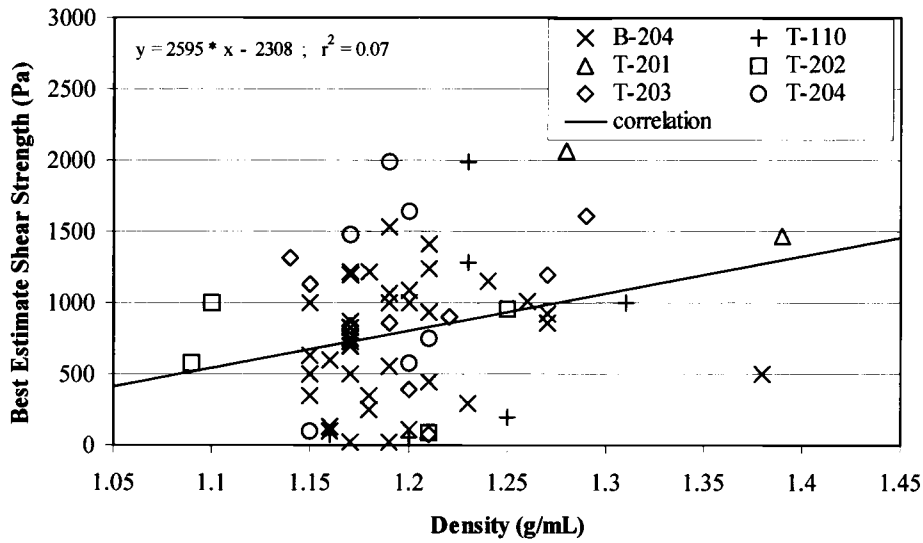


Figure 5.6. Relation of Estimated Waste Shear Strength and Bulk Density

6.0 Gas Retention

As noted in the introduction, the process of dry TRU waste retrieval may result in the entrainment of air (or other retrieval process gas) in waste transferred to a hopper for subsequent loading to drums. Experience with a variety of Hanford tank wastes suggests that some of this gas is likely to be trapped or retained within a fluffed sludge and not readily released. This in turn might negatively impact waste mass loading in drums. This section summarizes some earlier studies of gas generation and gas retention in Hanford tank wastes and simulants, emphasizing the potential for gas retention in TRU sludge.

Historically, several of the 177 waste storage tanks at Hanford are known to generate and retain flammable gases, as summarized by Johnson et al. (1997, 2001). Typically, the flammable gas watch-list tanks contain high-level wastes and significant quantities of organic material from complexants and other sources. The gas generated within the waste volume by thermal and radiolytic mechanisms sometimes accumulates as bubbles or voids, causing the waste volume in the tank to increase. The ability of various Hanford waste types to retain gas and the mechanisms of gas retention and release have been the focus of several studies at Hanford (e.g., Gauglitz et al. 1995, 1996; Gauglitz and Terrones 2002; Bredt et al. 1995; Bredt and Tingey 1996; Stewart et al. 1996a; Meyer et al. 1997; Rassat et al. 1997, 1998, 1999). In general, it was found that gas retention would occur in any tank (or vessel) that had a settled layer of wet solids provided there was sufficient gas generation. The retention of bubbles is not surprising and is known to occur in a variety of materials, ranging from yield stress fluids and pastes (Chhabra 1993) to ocean sediments (Wheeler 1990).

While the TRU waste tanks identified in this report have not posed a significant concern for generating and retaining flammable gas in situ, studies of Hanford wastes clearly indicate that TRU sludge waste could retain gas to a significant extent if gas were generated or added within a bulk quantity of the sludge. For example, Gauglitz et al. (1996) observed and measured retained gas in samples of Hanford tank sludge (Tank S-102) and in sludge simulants (bentonite clay/ water mixtures). They determined that the maximum volume fraction of gas retained in the bulk waste or simulant before being (partially) released from the sludge-like material is a function of strength. In a series of experiments with bentonite clay simulants of varying initial shear strength, the maximum retained gas void fraction peaked near 0.4 for relatively weak to moderate strength material (30–100 Pa shear strength). The maximum gas fraction in stronger bentonite mixtures was lower but still significant (~0.3 gas volume fraction at ~1000 Pa). Similar trends were observed in actual waste samples. The implication for TRU waste processing is that retrieved waste could retain significant volumes of gas (e.g., entrained air), depending on how the waste is handled and the strength of the material.

7.0 References

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Appendix

Development of Extrusion Length and Slump Methodologies to Estimate Yield Stress in Shear

Appendix

Development of Extrusion Length and Slump Methodologies to Estimate Yield Stress in Shear

A methodology developed by Gauglitz and Aikin (1997) allows the rheological properties of waste sediment (solid, liquid, and gas matrix) to be estimated based on a visual comparison of horizontal extrusion behavior for simulants with known yield stress in shear and that of Hanford waste. In Section A.1, the simulant extrusion behavior from Gauglitz and Aikin is re-analyzed to provide a methodology to estimate yield stress in shear from core extrusion lengths. A "slump" method for estimating the yield stress in shear of weaker materials is also presented. The results of these methodologies are compared with data for select Hanford wastes in Section A.2.

A.1 Horizontal Core Extrusion Yield Stress in Shear

Alternative methodologies for estimating the yield stress in shear from horizontal waste core extrusions are developed and presented from data provided in Gauglitz and Aikin (1997) and "slump" measurements described by Pashias et al. (1996). Gauglitz and Aikin horizontally extruded bentonite/water and kaolin/Ludox/water simulants of known yield stress in shear and recorded the extrusion characteristics. Of particular interest is the behavior of the material during the initial part of the extrusion. The initially extruded sample can be considered to act as a cantilever beam.

The maximum tensile stress in a round cantilever beam at the proportional limit (tensile yield stress) may be determined from

$$\sigma_{max} = A\frac{\rho g L^2}{d} \tag{A.1}$$

under the assumptions that

1. The beam (extruded waste or simulant sample) is of homogenous material that has the same modulus of elasticity in tension and compression.
2. The beam is essentially straight initially.
3. The cross section is uniform.
4. The beam has at least one longitudinal plane of symmetry.
5. All loads and reactions are perpendicular to a longitudinal plane of symmetry.
6. The beam is long in proportion to its depth.
7. The beam is not disproportionately wide.
8. The maximum stress does not exceed the proportional limit.

In Eq. (A.1), A is a constant, ρ is the density, g is the gravitational acceleration, L is the length of the beam supported at the proportional limit, and d is the diameter (for the basis of the equation, see Stevens 1979).

Gauglitz and Aikin modeled the initial horizontal extrusion of their simulants using an equation from Powell et al. (1995). The first term of their equation is equivalent to Eq. (A.1) (with A = 4.0) (Stevens 1979). A second term in the equation of Gauglitz and Aikin is an approximation of the maximum shear stress and accounts for approximately 4% (in the stronger simulants) to 30% (in the weaker simulants) of their calculated strength values.

The relationship of a material's yield stress in shear (τ_y) to its tensile yield stress is ambiguous. The relation is likely a function of the material microstructure (polycrystalline, polycrystalline face-centered cubic or body-centered cubic, etc.). The experimental results of Gauglitz et al. (1995) indicate that the gas content in a given material also affects this relationship. From their measured shear and tensile yield stresses in a bentonite clay simulant with the lowest gas content, σ_{max}/τ_y is approximately 2.0. They measured the yield stress in shear of the simulant using a shear vane, and the tensile yield stress was measured using a vertical extrusion technique. As the gas content was increased, the relationship of σ_{max}/τ_y was altered to the extent that, at gas concentrations of greater than approximately 10 to 15% by volume, the yield stress in shear was greater than the tensile yield stress. Gauglitz et al. noted that a "satisfying" explanation of the measured dependence of the tensile stress on the gas content has not been developed.

The shear and tensile yield stresses in a material are typically related through application of the von Mises ($\tau_y = \sigma_{max}/3^{0.5}$) or Tresca ($\tau_y = \sigma_{max}/2$) yield criterion. (Note the similarity of the lowest gas content results from Gauglitz et al. to the Tresca relation). The Taylor (or Taylor-Schmid) factor ($\tau_y = \sigma_{max}/3.06$) is also applied (Stroller and Zinkle 2000). By applying Eq. (A.1) with A= 4.0 to the horizontal extrusion results of Gauglitz and Aikin, σ_{max}/τ_y ranges from approximately 2 to 8. The stress ratio for the kaolin/Ludox simulant is inversely correlated with the measured yield stress in shear. The stress ratio varies from ~4 to 7 in the bentonite simulant, and no correlation is evident with the measured yield stress in shear. Regardless, acknowledging possible issues such as the preceding simulant results and those raised by the data in Gauglitz et al., theory suggests that, with the appropriate factor, f, the yield stress in shear of the horizontally extruded sample may be expressed from Eq. (A.1) as

$$\tau_y = \frac{A}{f}\frac{\rho g L^2}{d} \tag{A.2}$$

As is the norm in Hanford literature, the yield stress in shear is subsequently referred to in this section as the shear strength. As has been illustrated, uncertainty exists in the applicability and exact form of Eq. (A.1) as well as in the correct correlation of the tensile stress to the shear stress. However, Gauglitz and Aikin reported the shear strength and density of their simulants.

They also reported the length at which the extrusion exhibited "failure." With these data we can correlate the shear strength of the material directly with the functional form of Eq. (A.2) from

$$\tau_y = K\frac{\rho g L^2}{d} \tag{A.3}$$

The proportionality coefficient K of Eq. (A.3) provides a means to compute the shear strength of a material given its density and the plastic failure length of a horizontal extrusion. Plastic failure occurs in the relatively ductile bentonite simulant as necking, while the brittle kaolin/Ludox simulant fractures. The correlation from the bentonite clay simulant data of Gauglitz and Aikin is shown is Figure A.1. A strong correlation is identified. As discussed above, we expect from theory that the shear strength and $\rho g L^2/d$ for a given material will be linearly correlated. Additionally, for low shear strength, $\rho g L^2/d$ should approach zero (i.e., L goes to zero as τ_y goes to zero). An R^2 value of 0.98 is achieved with the intercept set to zero, and the proportionality coefficient (K) is 0.89.

The results for the kaolin/Ludox simulant are presented in Figure A.2. Given that the correlation is most likely a function of the material microstructure, a different correlation for the bentonite and kaolin/Ludox simulants is possible. With an intercept of zero (expected behavior as τ_y goes to zero), R^2 is 0.58 and the K is 1.45.

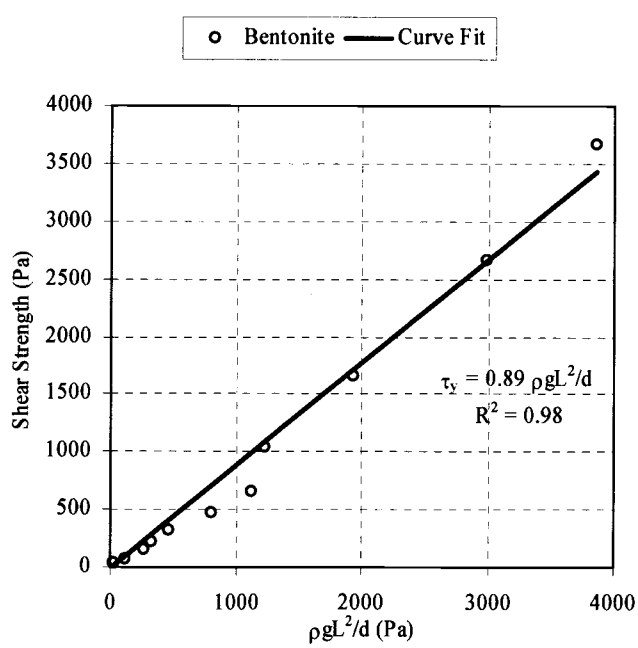

Figure A.1. Bentonite Simulant Correlation

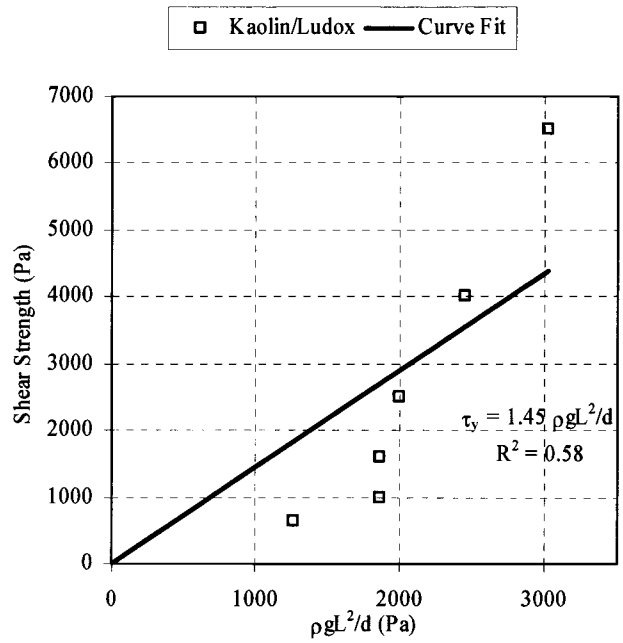

Figure A.2. Kaolin/Ludox Simulant Correlation

Based on this discussion, the shear strength of a material with a microstructure similar to the bentonite simulant is 0.89 times $\rho g L^2/d$. For materials with a microstructure similar to the kaolin/Ludox simulant, the shear strength is 1.45 times $\rho g L^2/d$. As reported in Gauglitz and Aikin (1997), the two simulants were chosen to reflect the wide variety of mechanical behaviors typical of wastes from the Hanford tanks. If we assume that the simulants bound the mechanical behavior of Hanford waste, we can expect that the shear strength of the waste will be between 0.89 and 1.45 times $\rho g L^2/d$. Recall, however, that a much better R^2 (0.98 to 0.56) was determined from the bentonite simulant data, so there is a higher level of confidence in the lower bound.

The methodology was back-applied to the simulant experiment horizontal extrusions. The K values of 0.89 and 1.45 are treated as bounding values and are subsequently referred to as the extrusion length bounds. The predicted shear strength calculated from the reported failure lengths (Gauglitz and Aiken 1997) and the measured shear strengths of the simulants are shown in Figure A.3. Because K values are applied indiscriminate of material type, it is of interest to consider the K that will provide the least error in predicted and actual yield stress in shear values for both simulants. From a least squares analysis applied to the entire group of simulants, the best fit is obtained with a K of 1.15. This fit is included in Figure A.3 as the "extrusion length best fit."

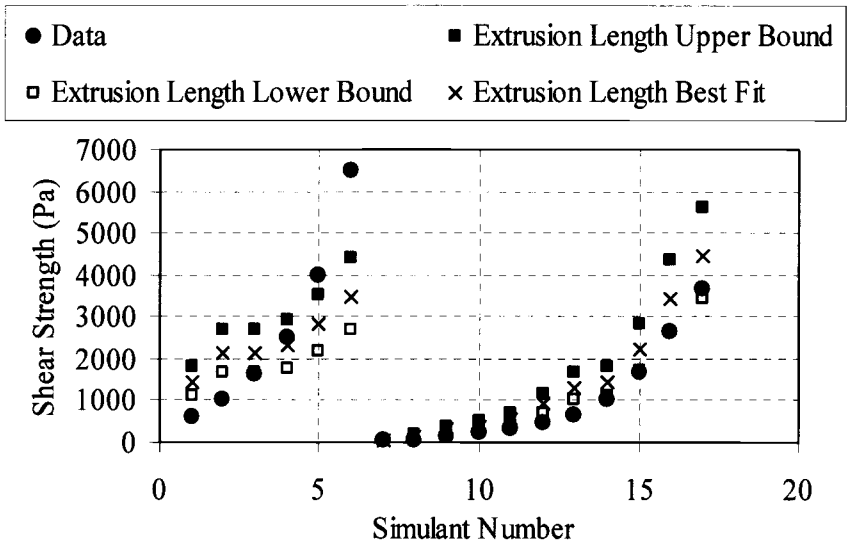

Figure A.3. Back-Application of Simulant Correlations to Simulants 1-6 Kaolin/Ludox; Simulants 7-17 Bentonite

If the extruded material is so weak that it "pours" out of the sampler instead of extending out, the yield stress in shear of the material may be determined by the amount of deformation, or slump, it undergoes. As depicted in Figure A.4, the slump is the change in the extrusion diameter. The slump is estimated from the core extrusions by comparing the diameter of the sampler with the height of the material after it has been extruded and is given by the slump length, s, divided by the extrusion diameter, d. The expression given by Pashias et al. (1996) for the yield stress as a function of slump in a vertical cylinder of sample was adapted to a horizontal cylinder. An expression for horizontal cylinders equivalent to the results of Pashias et al. for vertical cylinders was derived:

$$\tau_y = \frac{\rho g d}{2}\left[1-\left(\frac{s}{d}\right)^{\frac{1}{2}}\right] \tag{A.4}$$

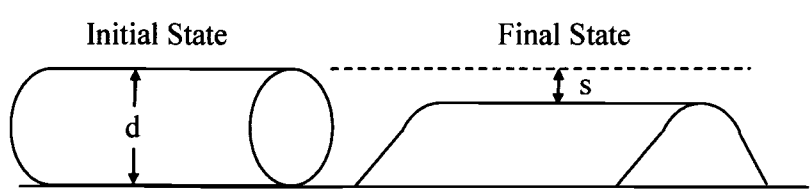

Figure A.4. Horizontal Cylinder Slump

The slump methodology has been shown to be an effective means to determine yield stress in shear of viscoplastic materials (Pashias et al. 1996; Schowalter and Christensen 1998). Slumping behavior in TRU waste ($\rho \approx 1.2$ g/mL) may be expected (from Eq. A.4 at 1% slump) to occur at approximately 150 Pa.

A.2 Application of Yield Stress Methodologies to Select DSTs

To investigate the applicability of these methodologies to tank waste, we compared the ball rheometer results (Stewart et al. 1996; Hedengren et al. 2000) with results achieved using the extrusion length and slump test in waste samples from double-shell tanks (DSTs) AN-103, AN-104, AW-101, and SY-103 for which extrusion videos were available and could be analyzed. The ball rheometer is deployed in situ, and the waste rheology is estimated directly from the drag force on the ball as it moves through the waste at various speeds. The ball rheometer results are typically accepted as being more representative of in situ waste conditions than laboratory rheological measurements (Hedengren et al.). Gauglitz and Aikin (1997) compared their visual results for AN-103, AW-101, and SY-103 extrusions to both the ball rheometer data and laboratory shear vane data. Their estimates generally agreed with the ball rheometer data within a factor of two.

Failure length measurements were taken from videos of the horizontal core extrusions from each waste. The recorded failure length was determined from the point at which plastic failure was judged to occur (i.e., the proportional limit was exceeded). The on-screen extrusion length is correlated to the actual length by measuring the on-screen extrusion diameter and correlating it to the actual extrusion diameter (1.125 in). This methodology forces a blind approach. Slump measurements, where appropriate, were determined by comparing the extrusion tube diameter with the extruded waste height. Given that the video footage was taken from outside a hot cell from various orientations, specific measurements may have significant uncertainty. In general, no shear strength estimates are reported for those samples where it would be difficult to discern the on-screen failure length to within approximately 10% of the measurement.

Data taken from the upper half of a core segment are ascribed to a level in the tank corresponding to 0.75 of the length of the segment plus the lower elevation of the segment. Similarly, data taken from the lower half of a core segment are assigned the segment elevation plus 0.25 of the segment length. Multiple measurements were available for individual segment halves in some instances, and each measurement is for reported for that elevation. Density values are taken from Hedengren et al. (2000).

The shear strength estimates for SY-103 using the extrusion length and slump methodologies are shown in Figure A.5. As presented in Section A.1, the extrusion length bounds are 0.89 and 1.45 times $\rho g L^2/d$, and the extrusion length best fit is 1.15 times $\rho g L^2/d$. Using the ball rheometer results as the benchmark, we see that results agree to within better than a factor of 2

for both slump and extrusion length estimates. In fact, the ball rheometer results are generally bounded by the extrusion length estimates.

Figure A.5 shows that the shear strength increases with depth in the SY-103 sediment layer. This phenomenon may be attributable to lithostatic loading and is therefore expected. The tank originally received concentrated complexant waste from B Plant via the 242-S evaporator. In late 1980, the tank was pumped down to a 46-in. "heel" and then received double-shell slurry and uranium sludge from ion exchange processing. Waste was also added from SX-104 (Stewart et al. 1996). The 46-in. heel is approximately equivalent to the height at which the ball rheometer was supported by the waste in both risers. When the ball is supported by the waste, the yield stress in shear is at least 900 Pa (Meyer et al. 1997).

As shown in the varied ball rheometer results for different risers, heterogeneities may be expected in the sediment layer. Any or all of the following may cause these heterogeneities: fill history, waste-disturbing activities (lancing, sampling, etc.), and/or spontaneous gas release. Thus, given a different riser or sampling time, the shear strength results are not necessarily expected to be in complete agreement.

The shear strength estimates for AW-101 are presented in Figure A.6. Again, results are generally within a factor of 2 or better of the ball rheometer measurements. The ball rheometer was supported at approximately 37 in. in riser 1C, almost twice the depth of the heel reached in 1986 (Stewart et al. 1996). The low slump estimate at approximately 20 in., though apparently

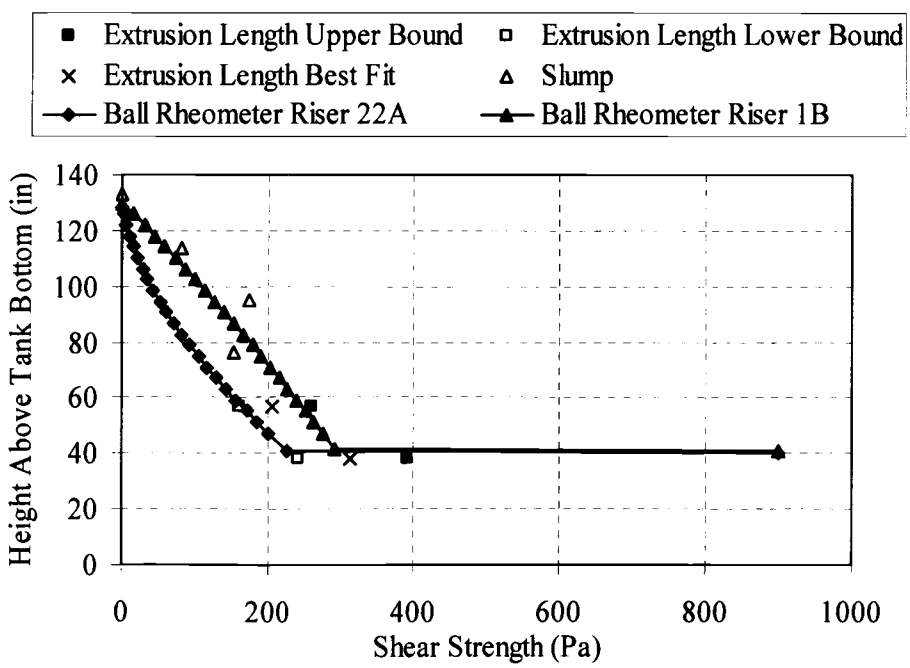

Figure A.5. Shear Strength as a Function of Height in SY-103

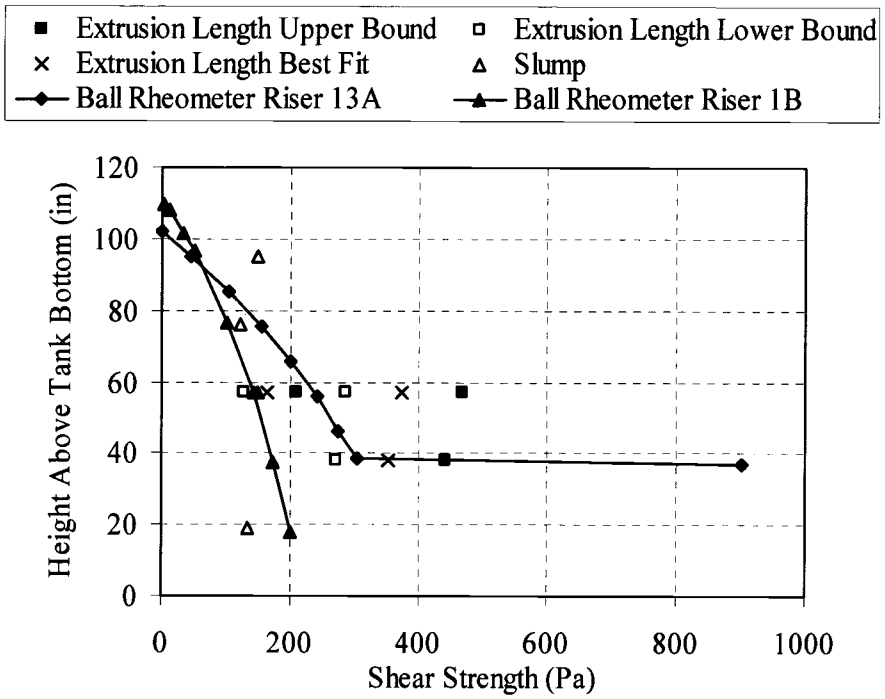

Figure A.6. Shear Strength as a Function of Height in AW-101

in agreement with the ball rheometer in riser 1B, results from significantly more fluid waste at the top of the sampler than that surrounding (above and below) it. It is hypothesized that this is a result of the sampling/handling process and is therefore not representative of in situ conditions.

Similar results are attained for AN-104 (Figure A.7) and AN-103 (Figure A.8). The 900-Pa values from the ball rheometer are reported where downward motion of the ball was prevented by the waste. Low slump values deep in the sediment are similar to AW-101 and are significantly more fluid than the surrounding waste.

The extrusion length and slump methodology results are similar in magnitude and reproduce the same trends as the ball rheometer results. In the absence of definitive in situ measurements, or in support of them, the methodologies applied here to horizontal waste core extrusion behavior are expected to produce representative results for the shear strength.

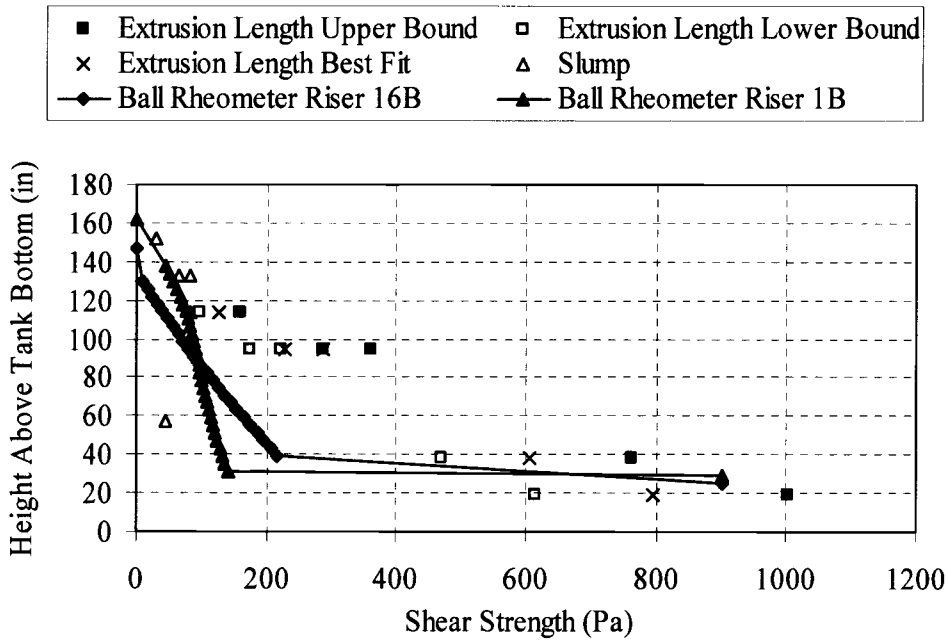

Figure A.7. Shear Strength in Shear as a Function of Height in AN-104

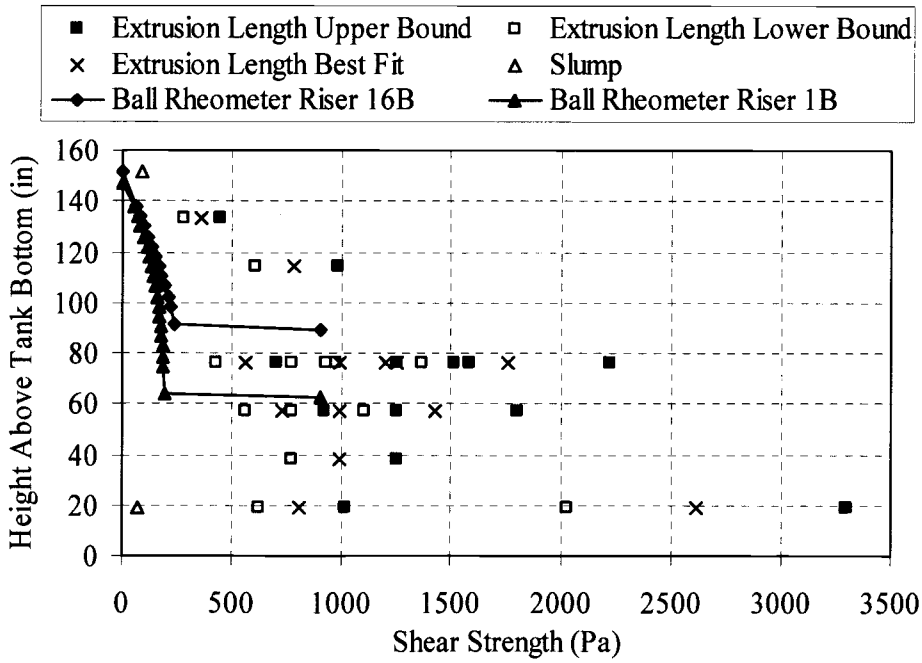

Figure A.8. Shear Strength as a Function of Height in AN-103

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Dangerous Waste Characteristics of Contract-Hanford Transuranic Mixed Wastes from Hanford Tanks

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J.R. Deschane

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Dangerous Waste Characteristics of Contact-Handled Transuranic Mixed Wastes from Hanford Tanks

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Executive Summary

Existing analytical data from samples taken from Hanford tanks designated as potentially containing transuranic mixed process wastes, along with process knowledge of the wastes transferred to these tanks, have been reviewed to determine whether the dangerous waste characteristics currently assigned to all waste in Hanford underground storage tanks are applicable to these wastes. Supplemental technologies are being examined to accelerate the Hanford tank waste cleanup mission and to treat the waste in the safest and most efficient way. To date, 11 Hanford tanks have been designated as potentially containing contact-handled (CH) transuranic mixed (TRUM) wastes. The CH-TRUM wastes are found in single-shell tanks (SSTs) B-201 through B-204, T-201 through T-204, T-104, T-110, and T-111. Methods and equipment to solidify and package the CH-TRUM wastes are part of the supplemental technologies being evaluated. The resulting packages and wastes must be acceptable for disposal at the Waste Isolation Pilot Plant (WIPP).

The 11 tanks reviewed were categorized into three groups based on their process history and the Best Basis Inventory. These three categories are T Farm 100 series tanks receiving first-cycle decontamination waste from the $BiPO_4$ process, T Farm 100 series tanks receiving second-cycle waste from the $BiPO_4$ process and lanthanum fluoride finishing waste, and B- and T-200 series tanks receiving lanthanum fluoride finishing waste.

The dangerous waste characteristics being considered include ignitability (D001), corrosivity (D002), reactivity (D003), and toxicity arising from the presence of 2,4,5-trichlorophenol at levels above the dangerous waste threshold (D041). The analytical data reviewed with respect to waste code D001 included differential scanning calorimetry (DSC) and the percent of the lower flammability limit (LFL) calculated from the composition of the headspace. Concentrations of sulfur, sulfate, cyanide, and the composition of headspace (vapor space) and DSC were reviewed for waste code D003; and pH was reviewed for D002. For waste code D041, the analytical data reviewed included the concentrations of 2,4,5-trichlorophenol, total organic carbon (TOC), and oxalate. Data on mercury concentrations in these wastes were also included. DSC results were used to determine the energetics of the tank wastes as a function of temperature. These results are summarized for each of the three tank categories.

No exothermic behavior was observed in the majority of the CH-TRUM wastes. Exothermic transitions were observed in a limited number of samples from Tank B-203, and a significant number of samples showed consistent exothermic behavior in Tanks B-202 and T-111. Exothermic transitions observed in these samples were generally broad peaks with small amplitudes at temperatures exceeding 200°C, indicating that explosive reactions or ignition hazards at standard temperatures or pressures are unlikely (waste codes D001 and D003). The waste with the greatest potential for exothermic reactions is T-111. The reactions are most likely due to the oxidation of organic compounds that are found primarily in the top section of the waste. If the wastes are wet, significant energy must be expended to remove the water from the waste before any reaction can occur.

Sulfur and sulfate analyses indicated that the majority of the sulfur in the waste is present as sulfate. Based on the pH of the tank wastes, the sulfate is stable and will not react to form sulfide; therefore, the CH-TRUM wastes are not sulfide-bearing wastes (waste code D003). A limited number of cyanide

analyses were available for these tanks as well. Results indicated that these wastes do not contain a sufficient concentration of cyanide to be considered cyanide-bearing wastes (waste code D003).

Concentrations of the 2,4,5-trichlorophenol measured in the CH-TRUM wastes did not exceed the dangerous waste limits; therefore, the wastes do not meet the criteria of toxic characteristics based on this organic constituent (waste code D041). Oxalate and TOC data indicate that the majority of the organic carbon in the waste exists as oxalate.

The pH of the CH-TRUM wastes is too low to satisfy the pH criterion of a corrosive waste (waste code D002). Gas analysis of the headspace vapors in these tanks indicates that all of the toxic vapors except ammonia are well below the threshold limit value-time weighed average (TLV-TWA) (waste code D003). The tank wastes will need to be diluted prior to retrieval or other mitigation methods to meet this ammonia criterion on all of the wastes except those in the B- and T-200 series tanks.

Analysis of the existing characterization data from core, grab, and vapor samples from the CH-TRUM wastes supports the removal of the dangerous waste codes for ignitability (D001), corrosivity (D002), and toxicity arising from the presence of 2,4,5-trichlorophenol (D041). For the B- and T-200 series tanks, the data also support the removal of the dangerous waste code for reactivity (D003). All constituents and properties were below the levels required to remove waste code D003 from Tanks T-104, T-110, and T-111 except ammonia, which exceeds the TLV in the headspaces of these tanks.

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1.0 Introduction

Several dangerous characteristics have been assigned to the wastes found in the Hanford underground storage tanks. These characteristics include ignitability, corrosivity, reactivity, and toxicity arising from the presence of 2,4,5-trichlorophenol at levels above the dangerous waste threshold. Existing analytical data from samples taken from the Hanford tanks designated as potentially containing contact-handled (CH) transuranic mixed (TRUM) process wastes, along with process knowledge of the wastes transferred to these tanks, have been reviewed to determine whether the waste codes are applicable to these tank wastes.

1.1 Background

Radioactive wastes from defense operations on the Hanford Site were accumulated in 177 underground storage tanks beginning in the 1940s. These wastes came from three different processes for recovering uranium and plutonium from irradiated uranium fuel, three different processes for recovering radionuclides from the waste, from miscellaneous other sources such as laboratories and reactor decontamination solutions, and from production and waste management operations. The acid waste streams were made pH neutral or alkaline before being transferred into the tanks and formed metal hydroxide sludges. Evaporation of water from these wastes concentrated the wastes to form crystallized salts and salt-rich alkaline solutions.

The U.S. Department of Energy Office of River Protection (DOE-ORP) has responsibility for managing the safe storage, treatment, and disposal of this waste. The tanks contain approximately 53 million gallons of highly radioactive waste and approximately 190 million curies of radioactivity. Current disposition strategies for the majority of these waste includes retrieval of the wastes from the tanks, pretreatment, and vitrification. A Waste Treatment Plant (WTP) is currently being designed and built to perform these operations.

Supplemental technologies are being examined to accelerate the Hanford tank waste cleanup mission and to accomplish waste treatment in a safer and more efficient manner. Methods and equipment to solidify and package the CH-TRUM wastes contained in 11 single-shell tanks (SSTs) are part of the supplemental technologies being evaluated. The resulting packages and wastes must be acceptable for disposal at the Waste Isolation Pilot Plant (WIPP).

1.2 Transuranic Mixed Wastes (TRUM)

To date, 11 Hanford waste tanks have been designated as potentially containing CH-TRUM wastes. The CH-TRUM wastes are found in SSTs B-201 through B-204, T-201 through T-204, T-104, T-110, and T-111. Waste from Tank T-110 was initially classified as low-level waste but is being considered a candidate for designation as potentially containing TRU process waste (Gasper et al. 2002).[a]

(a) The proposed TRUM waste packaging system will dry the Tank T-110 waste and increase the concentration of TRU to greater than 100 nCi/g. Thus, the waste in Tank T-110 will meet the minimum TRU concentration requirement for CH waste disposal at the WIPP.

A modified dry-retrieval process is proposed to remove the TRUM waste from the Hanford tanks. The sludge is removed from the tank using a vacuum retrieval system. An aqueous stream flowing at approximately 1 to 5 gpm will be added to the retrieved waste in the vacuum line to transfer the diluted waste to a storage vessel. A significant fraction of the liquid in this diluted waste stream must be removed to meet the WIPP disposal criteria. A vacuum drying method (at less than 70°C) has been selected for removing water from this diluted waste. An absorbent may be added to the waste package to prevent the formation of free liquid during handling, transport, and storage of the package.

1.3 Dangerous Waste Characteristics

Dangerous waste characteristics, as described in the Dangerous Waste Regulations published by the Washington State Department of Ecology (Ecology 2003), that are assigned to these tanks and are of interest to this study include ignitability, corrosivity, reactivity, and toxicity. Other dangerous waste characteristics are assigned to the TRUM wastes but were not included in this analysis. Characteristics of ignitability (D001), corrosivity (D002), reactivity (D003), and toxicity from 2,4,5-trichlorophenol (D041) associated with the TRUM wastes were considered.

Hanford tank wastes exhibit the characteristics of ignitability if a representative sample of the waste is "capable, under standard temperature and pressure, of causing a fire through friction, absorption of moisture or spontaneous chemical changes and, when ignited, burns so vigorously and persistently that it creates a hazard" or "it is an oxidizer." The other criteria described in the Dangerous Waste Regulations do not apply because the liquids associated with the tank waste are aqueous solutions containing less than 24 percent alcohol by volume, and the tank wastes are not compressed gases.

Wastes that have a pH less than or equal to 2 or greater than or equal to 12.5 are designated as dangerous wastes due to corrosivity and assigned the dangerous waste number of D002 or WSC2. A designation of WSC2 is assigned when the pH of the liquid generated from mixing solid or semisolid waste with equal amounts of water is less than or equal to 2 and greater than or equal to 12.5. If the pH measured directly on the waste is less than or equal to 2 and greater than or equal to 12.5, the waste is corrosive and is assigned the dangerous waste designation D002.

Several waste properties must be considered to determine whether it exhibits reactivity. These properties include a representative sample of the waste (1) being normally unstable and readily undergoing a violent change without detonation; (2) reacting violently with water; (3) forming potentially explosive mixtures with water; (4) generating toxic gases, fumes, or vapors at concentrations sufficient to present a danger to human health or the environment when mixed with water; (5) containing cyanide or sulfide which when exposed to pH conditions between 2 and 12.5 can generate toxic gases, vapors, or fumes at concentrations sufficient to present a danger to human health or the environment; (6) being readily capable of detonation or explosive reaction if it is subjected to a strong initiating source or if heated under confinement; (7) being readily capable of detonation or explosive decomposition or reaction at standard temperature and pressure; and (8) being a forbidden explosive. These properties deal primarily with the energetics of the waste or generation of toxic gases from the waste.

Toxicity characteristics are based on concentrations of contaminants at or above the dangerous waste thresholds. Only a single contaminant (2,4,5-trichlorophenol) on the toxicity characteristics list is being considered for these tank wastes. The dangerous waste threshold for 2,4,5-trichlorophenol is 400 mg/L.

2.0 Tank Waste History Summaries

Between 1943 and 1964, 149 SSTs were built for storing radioactive wastes generated by the chemical processing of irradiated reactor fuels. The SSTs are located in 12 tank farms in the 200 West and 200 East Areas on the Hanford Site. Figure 2.1 is a reference schematic of these SST farms and the associated six double-shell tank (DST) farms. The capacities of the SSTs range from 208 m^3 (55,000 gallons) to 3,785 m^3 (1,000,000 gallons). Carbon steel lines the bottom and sides of the reinforced concrete shell of each tank. The tanks are below grade with at least 6 feet of soil covering them. A sketch of a typical SST is provided in Figure 2.2.

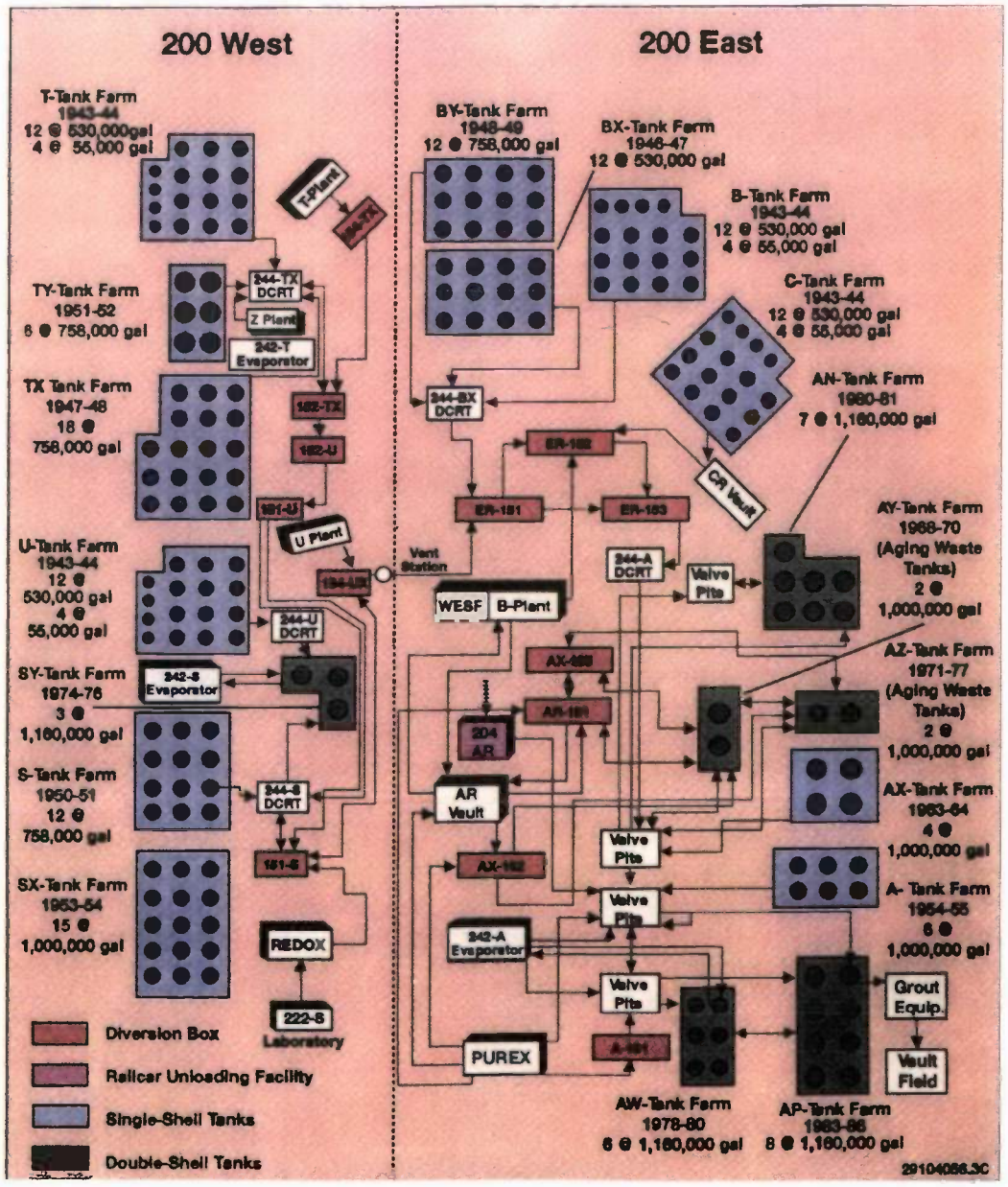

Figure 2.1. Hanford Site Tank Farms

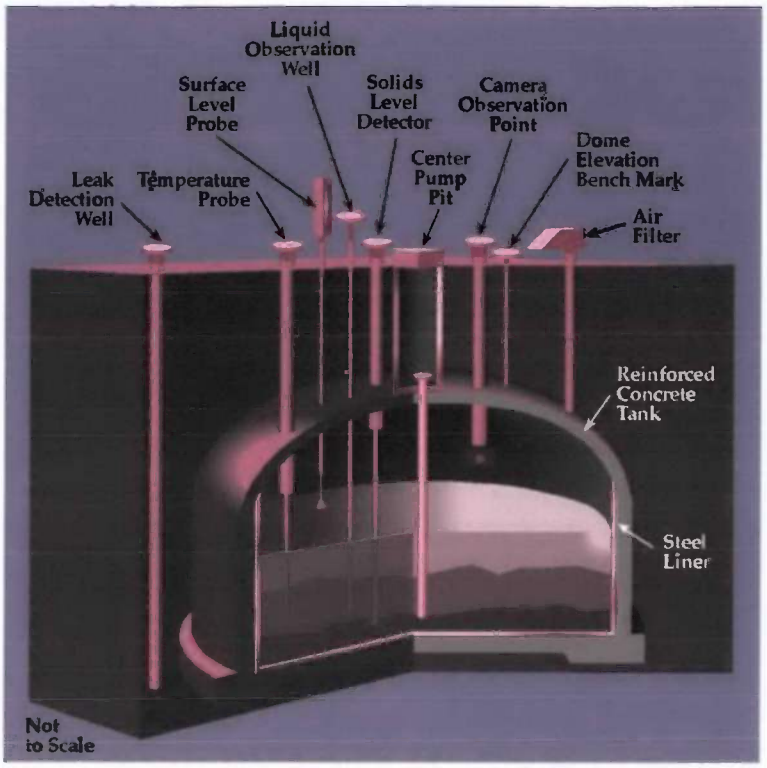

Figure 2.2. Typical Single-Shell Tank

Many of the SSTs were built in "cascades" of three, four, or six tanks. Inlet and overflow lines in these tanks are near the top of the carbon-steel liner. Waste was transferred to the first tank of the cascade and allowed to overflow into the successive tanks of the cascade through the overflow lines.

Twenty-eight DSTs were constructed between 1968 and 1986 to receive liquid radioactive wastes generated by decommissioning and cleanup operations as well as irradiated fuel processing in the 100, 200, 300, and 400 Areas of the Hanford Site. These operations included the transfer of pumpable liquids from the SSTs to the DSTs. Each DST consists of three concentric structures. The outer tank structure is a reinforced concrete tank that is lined with a secondary carbon steel liner that extends along the concrete tank haunch and dome to the inner tank haunch. The inner structure is a free-standing, completely enclosed carbon steel tank located within the secondary liner and separated by an annular space. Leak detection and liquid level detection devices are placed in the annular space. Figure 2.3 is a sketch of a typical DST.

Access to the waste inside the tank is provided by risers that penetrate the tank's dome. The risers vary in diameter from 4 to 42 inches, and the number and size of the risers vary from tank to tank. Both sampling and monitoring of the tanks are performed through these risers, and many of them have been filled with monitoring instrumentation, limiting the locations at which the tank can be sampled.

Several methods have been used at the tank farms to obtain samples of the tank waste. The primary sampling methods include core sampling, grab sampling (also called Bottle-on-a-String), vapor space (headspace) sampling, and auger sampling.

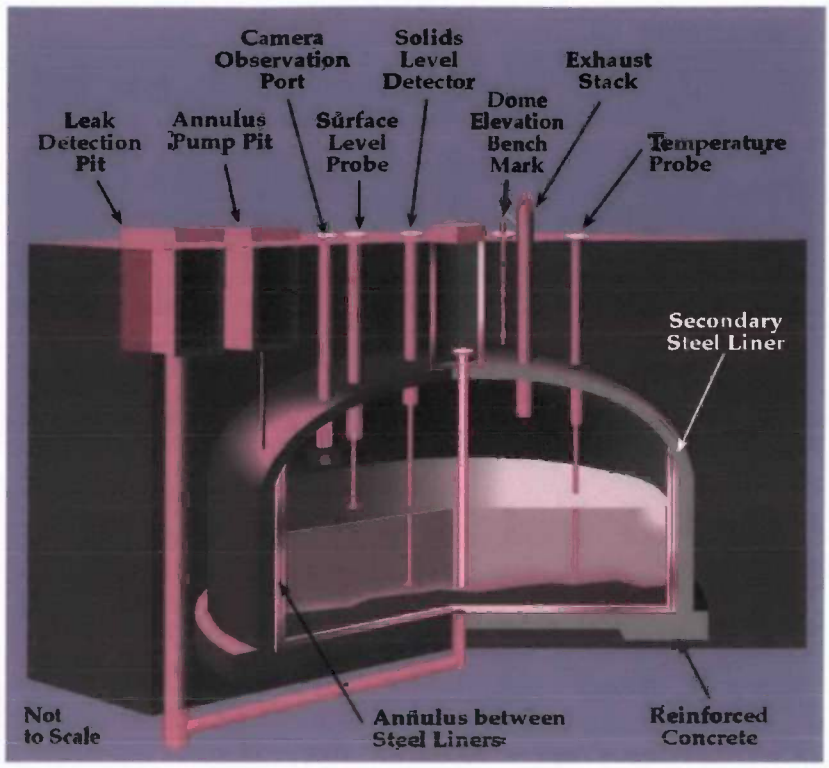

Figure 2.3. Typical Double-Shell Tank

Core samples are solid or liquid samples taken in 19-inch segments throughout the depth of the tank. Core samples are obtained from the tanks using a specially designed core drilling truck and sampling device that is either pushed or rotated through the waste. Core samples provide data on the variation in the composition and properties of the waste as a function of depth in the tank.

Grab samples are liquid or soft slurry samples that are taken from various depths in the liquid waste. A stoppered bottle is lowered to the desired depth in the tank, and the stopper is removed. The bottle fills with the liquid or slurry and is retrieved from the tank.

Vapor space or headspace samples are gas samples obtained at various heights above the surface of the waste in the tank. The vapor is collected in sorbent tubes, SUMMA canisters, or a cryogenic trap. The samples are analyzed in the laboratory by gas chromatography with a mass spectrometer as the detector (GC/MS) or by a gas mass spectrometer. Key analytes are reported along with any other gases that are observed in the samples.

2.1 241-B-201 through B-204 and 241-T-201 through T-204

The B-200 and T-200 series tanks are SSTs that were constructed between 1943 and 1944. The B and T tank farms are in the 200 East and 200 West areas, respectively. Each tank farm contains 12 100-series tanks and four 200-series tanks (Brevick et al. 1997). The 200-series tanks have a design capacity of 208 kL (55 kgal). Some reports list the design capacity of the B 200-series tanks as 204 kL (54 kgal) but

list the same diameter and operating depth. The reported diameter and operating depth at overflow for these tanks is 6.1 m (20 ft) and 7.49 m (24.58 ft), respectively. The tanks are passively ventilated. Tanks B-201 and T-201 were piped separately from the other tanks; whereas the B-202, B-203, and B-204 tanks were connected via tie lines, and the T-202, T-203, and T-204 tanks were connected via tie lines. Waste was cascaded through the interconnected tanks. The B-200 and T-200 series tanks received waste from 224-B and 224-T Concentration Buildings, respectively.

The waste in SSTs B-201 through B-204 and T-201 through T-204 was initially produced in the 224-B or 224-T Concentration Building operation as part of the plutonium concentration cycle in the bismuth phosphate process (DuPont 1944). According to the 7/2004 Best-Basis Inventory Derivation Reports, Tanks B-201 and T-201 received lanthanum fluoride finishing waste produced before 1949 (224-1), and the remaining B- and T-200 series tanks received lanthanum fluoride finishing waste produced after 1949 (224-2).

Lanthanum fluoride was added as a carrier precipitate to the plutonium product to remove bismuth and fission products not completely scavenged by the bismuth phosphate in the processing steps that were conducted in the 221-B and 221-T Plants. The 224-B/224-T Concentration Building waste comprised both solid and supernatant fractions. The solids were settled in these tanks (Gasper et al. 2002; Anderson 1990), and the bulk of the fission products settled out with precipitated phosphates and lanthanum fluoride. Fission-product activity of the supernatant fraction of these wastes was low enough (<0.001% of the activity in the source material) to permit ground disposal.

Tank B-201 went into service in October 1946 and stopped receiving waste from the 224-B Concentration Building in October 1948. Tanks B-204, B-203 and B-202 were then used from October 1948 through September 1952 to receive waste from the 224-B Concentration Building. The 221-B Plant and 224-B Concentration Building were shut down in September 1952, and the process equipment was flushed between October 1952 and March 1953, with other SSTs also receiving flush solutions from the 221-B Plant. Tanks B-201 through B-204 received flush water periodically from October 1954 through September 1955. A final batch of flush water was added to these tanks in the first half of 1962.

Tank T-201 went into service in November 1946 and stopped receiving waste from the 224-T Concentration Building in May 1949. Tanks T-204, T-203 and T-202 were then used from late May 1949 through May 1952 to receive waste from the 224-T Concentration Building (Johnson 2003). These tanks were declared inactive in 1977. Interim stabilization for all tanks was completed between April 1981 and June 1984, and intrusion prevention was completed between May 1981 and June 1985. The integrity of the T-200 series tanks and Tank B-202 is sound. Tanks B-201, B-203, and B-204 are assumed to be leakers; that is, to have previously leaked waste to the ground.

All of the tanks contain sludge, and Tanks B-203, B-204, and T-201 also contain liquid. Total tank volumes range from 21,000 to 52,000 gallons with the sludge making up most of that volume (21,000 to 51,000 gallons). The liquid volumes were a small fraction of the total volume in the tanks (ranging from 1,000 to 2,000 gallons in the three tanks that had liquid). The volumes for each tank are reported in Table 2.1.

Table 2.1. Waste Volumes in the T- and B-200 Series Tanks as of April 2004

Tank	Waste Volume kL (kgal)		
	Sludge	Supernatant	Total
B-201	115 (30	0	115(30)
B-202	111 (29)	0	111 (29)
B-203	193 (51)	2 (1)	195 (52)
B-204	189 (50)	3 (1)	192 (51)
T-201	110 (29)	9 (2)	119 (31)
T-202	81 (21)	0	81 (21)
T-203	140 (37)	0	140 (37)
T-204	143 (38)	0	143 (38)

All samples taken from the B-200 and T-200 series tanks were obtained by core sampling in the push mode. Full cores were obtained for all tanks except B-203 core 115, which omitted the last ~5 feet. Two additional cores (120 and 122) were obtained from Tank B-203 to provide a complete profile of the waste in this tank (see Figure A.1).

Core profiles are included in the appendix for all tanks except B-201 and B-202. These profiles are single-page diagrams with physical descriptions of the core sampling events for one tank. Core profiles were not available for Tanks B-201 and B-202, but characterization reports provide enough data to determine that full core samples were obtained for both (Heasler et al. 1994; WHC 1995). Eight segments were obtained from cores 26 and 27 from Tank B-201 and seven and eight from cores 24 and 25, respectively, from Tank B-202. No solids and only 90 mL of drainable liquid were obtained in the first segment of core 25. The other core segments were approximately 19 inches in length; thus, approximately 12.7 and 11.1 feet of waste depth were sampled in cores 26 and 27 and 24 and 25, respectively. Because the waste height is approximately 12.77 ft in B-201 and 12.3 ft in B-202, full cores were obtained from both tanks. The data and core identification for each sampling event are listed in Table 2.2.

Table 2.2. Sampling Events in the B- and T-200 Series Tanks

Tank	Core Samples		
	Core Number	Date Sampled	Riser
B-201	26	July 1991	2
	27	July 1991	7
B-202	24	June 1991	2
	25	July 1991	5
B-203	115	November 1995	2
	120	December 1995	2
	122	December 1995	7
B-204	112	October 1995	2
	114	October 1995	7
T-201	192	April 1997	3
T-202	191	April 1997	3
T-203	190	April 1997	3
T-204	188	March 1997	3

2.2 241-T-110 and 241-T-111

Tanks T-110 and T-111 are 2,006,000-L (530,000-gal) SSTs in the T tank farm (see Figure 2.1). Tank T-110 was designed for non-boiling waste with a maximum fluid temperature of 104°C (219°F). This tank is the first in the line of cascading tanks (T-110, T-111, and T-112), each one a foot lower in elevation than the preceding tank. A description of the tanks plus the tank volumes and sampling events are provided in Tables 2.3 and 2.4.

The wastes received in Tanks T-110 and T-111 are similar to those in the T-200 series tanks except that Tanks T-110 and T-111 also received waste originating from the second plutonium decontamination cycle (2C) from the 221-T Plant (DuPont 1944). Tank T-110 first began receiving second-cycle waste from 221-T Plant in December 1944. Additional second cycle waste was added from 221-T Plant in the first quarter of 1948 until the third quarter of 1956. From the second quarter of 1952 to the first quarter of 1953, the tank also received lanthanum fluoride waste from the 224-T Plutonium Concentration Building. Flush water was added in the second and third quarters of 1956 before the tank was placed on standby.

Table 2.3. 241-T-110 Tank Description and Status

Tank Description	
Type	Single-shell
Constructed	1943-1944
In-service	12/1944
Diameter	22.9 m (75 ft)
Operating Depth	469.9 cm (185 in)
Design Capacity	2006 kL (530 kgal)
Bottom shape	dish
Ventilation	Passive
Tank Status (as of 4/1/2004)	
Total Waste Volume	1400 kL (370 kgal)
Supernatant Volume	3 kL (1 kgal)
Sludge Volume	1397 kL (369 kgal)
Surface Level (4/1/2004)	370.8 cm (146.0 inches)
PCSACS Surface Level (8/16/2004)	370.7 cm (145.9 inches)
Integrity	Sound
Waste Group Designation	B
Sampling Dates	
Core Samples	1/29/1996 - 2/10/1996
Grab Samples	1/8/1997 - 1/8/1997
Vapor Samples	8/31/1995
Service Status	
Declared Inactive	1976
Interim Stabilization	January 2000

Table 2.4. 241-T-111 Tank Description and Status

Tank Description	
Type	Single-shell
Constructed	1943-1944
In-service	10/1945
Diameter	22.9 m (75 ft)
Operating Depth	469.9 cm (185 in)
Design Capacity	2006 kL (530 kgal)
Bottom shape	dish
Ventilation	Passive
Tank Status (as of 4/1/2004)	
Total Waste Volume	1691 kL (447 kgal)
Sludge Volume	1691 kL (447 kgal)
Surface Level (4/1/2004)	432.3 cm (170.2 inches)
PCSACS Surface Level (8/16/2004)	432.6 cm (170.3 inches)
Integrity	Assumed leaker
Waste Group Designation	B
Sampling Dates	
Core Samples	10/22/1991 - 10/25/1991
	11/5/1991 - 11/7/1991
Vapor Samples	1/20/1995
Service Status	
Declared Inactive	1976
Interim Stabilization	February 1995
Intrusion Prevention	

The tank was inactive and full until the second quarter of 1974, when supernatant waste was sent to Tank S-110. Supernatant waste was sent to T-101 in the first and second quarters of 1976 and to TX-118 in the third quarter of 1978. The last transfer from Tank T-110 prior to core sampling of the tank was saltwell pumping of liquids to Tank AN-103 in the fourth quarter of 1983. Saltwell pumping began again in May 1997, and the pump remains within the waste. As of August 31, 1997, 63.2 kL of supernatant liquid had been pumped from the tank, including the sluice water and any liquids that may have intruded into the tank from outside via pump pits.

Tank T-111 was brought into service during the fourth quarter of 1945 with a cascade of second cycle decontamination (2C) waste from Tank T-110 (Agnew et al. 1997b). The tank was filled with 2C waste, and the waste was cascaded to Tank T-112. Cascading continued until the third quarter of 1946, when Tank T-112 was filled. During the third and fourth quarters of 1947, nearly all of the supernatant of Tank T-111 was transferred to crib T-006. The cascading of 2C waste resumed in the first quarter of 1948. When the entire cascade became full, waste from T-112 was transferred to a crib. This cycle continued until the fourth quarter of 1952. From 1952 to 1956, Tank T-111 was used to cascade 2C waste from 221-T Plant and lanthanum fluoride waste (224) from the lanthanum fluoride finishing process in the 224-T plutonium concentration building to Tank T-112, which discharged to a crib. In 1995, supernatant waste was transferred from the tank to crib T-005.

Tank T-111 contents remained unchanged until the second quarter of 1974. From 1974 to 1976, 63,000 gallons of supernatant were transferred to Tanks S-110, T-101, T-109, and TX-109. Tank T-111 was declared inactive after these transfers were complete in 1976. Saltwell liquid was pumped from the tank in support of tank stabilization efforts in the fourth quarter of 1990, the fourth quarter of 1994, and the first quarter of 1995. The tank is currently characterized as an assumed leaker (Hanlon 2002).

The two cascade tanks (T-111 and T-112) from Tank T-110, along with Tank T-110, have been sampled and analyzed, and the resulting data contribute significantly to the understanding of Tank T-110. Bismuth phosphate second-cycle (2C) waste was the principal waste type added to T-110. For a short period of time, lanthanum fluoride finishing (224) waste was added to the tank and cascaded over into Tanks T-111 and T-112. Predicting the relative proportions of 2C waste solids and 224 waste solids to the overall tank waste inventory is difficult.

Due to the limited analytical information on the solids from the 1996 samples, estimates of tank contents were made based on data for Tanks T-111 and T-112, which indicate a higher contribution of lanthanum fluoride finishing (224) waste from Tank T-110, with a 2C/224 waste ratio of 80/20 or 75/25. Other tanks with sample data for the lanthanum fluoride finishing waste type include the T-200 and B-200 series tanks. Tanks B-110, B-111, and B-112 also contain a waste layer that is representative of 2C waste.

All appropriate data quality objectives (DQO) and waste issues have been addressed for Tank T-110 and accepted by the Project Hanford Management Contract Tank Waste Remediation System Program. No additional sampling and analyses are needed to satisfy current safety issue requirements for this tank. Tank T-110 was core-sampled through two risers between January 29 and February 7, 1996. Core 180 was taken from riser 6 on February 6 and 7, 1996, and core 181 was taken from riser 2 on January 29 and 30, 1996.

All applicable DQOs and waste issues have been addressed for Tank T-111, and no additional sampling and analyses are necessary to satisfy current safety issue requirements. Tank T-111 was push-mode core-sampled through three risers between October 22 and November 7, 1991. Core 31 was taken from riser 6 on October 22, 1991, and core 32 was taken from riser 2 on October 24 and 25, 1991. Core 33 was taken from riser 3 between November 5 and 7, 1991.

A hydrostatic fluid of normal paraffin hydrocarbons (NPH), similar to kerosene, was used in establishing a head balance while taking push-mode cores. Objections involving sample degradation and contamination were raised regarding the use of this fluid, and the practice has since been discontinued. For cores 31 and 33, nearly full recovery was achieved in every case.

The casks were transported to the 222-S Laboratory for characterization. Some of the physical tests, organic analyses, and uranium and plutonium isotopic analyses were performed at the Radiochemical Processing Laboratory (RPL) operated by Pacific Northwest National Laboratory (PNNL).

For cores 31 and 33, sample recovery was excellent; overall recoveries were in excess of 80 percent. Segment recoveries were based on the maximum recoverable volume for the segment regardless of solid/liquid ratio. The core recoveries reported in the data package were determined based on a visual inspection of the sample length and apparent volume at the time the samples were extruded. Although

samples for core 32 were taken from riser 2, the materials obtained at all levels appeared to be particulate suspended in an aqueous solution, with slight traces of NPH contamination observed in a few samples. These samples did not correspond to the observed conditions in the tank and were considered nonrepresentative. The results of the core 32 sampling exercise were attributed to sampler failure. Because no acceptable samples were acquired, no assays were performed on core 32 and no results were reported. The analytical data were reported in McKinney et al. (1993).

Grab samples were obtained from Tank T-111 on March 5, 1994.[(a)] Three 100-mL supernatant samples were retrieved from riser 13 (saltwell screen) in accordance with waste compatibility program requirements (WHC 1994). The compatibility samples were taken for emergency pumping of T-111 to SY-102. The samples were sent to the 222-S laboratory for analysis on March 25, 1994. Quality control (QC) analyses were not conducted for the three grab samples.

Vapor sampling of Tank T-111 to support the Health and Safety DQO (Osborne and Buckley 1995) was performed on January 20, 1995 using the vapor sampling system (VSS). Air from the T-111 headspace was withdrawn via a 6.1-m (20-ft) long heated sampling probe mounted in riser 3 and transferred through heated tubing to the VSS sampling manifold. All heated zones of the VSS were maintained at approximately 50°C (120°F) (Huckaby and Bratzel 1995). Samples were collected in SUMMA™ canisters or various types of sorbent traps. Samples collected in a triple sorbent trap device were analyzed by Oak Ridge National Laboratory (ORNL) for organic vapors. PNNL analyzed both SUMMA and sorbent trap devices for inorganic and organic vapors. Because of differences in documenting quality assurance measures between ORNL and PNNL, PNNL SUMMA sample results should be considered the primary organic vapor data for T-111.

The Best Basis Inventory (BBI) for T-111 and T-110 incorporates waste-type templates that correlate with the waste types in the tank. Templates are based on sampling data from tanks that contain the same waste type as Tanks T-111 and T-110, supplemented with Hanford Defined Waste (HDW) model (Agnew et al. 1997a) data and contribute significantly to the understanding of the concentrations of certain constituents in these tanks. The BBI source data are provided in Tables 2.5 and 2.6. The sludge waste phase includes both solids and interstitial liquid.

Table 2.5. Tank 241-T-110 Best Basis Inventory Source Data

Waste Phase	Waste Type	Associated Volume
Supernatant	2C1	3 kL (1 kgal)
Sludge	224-2	37 kL (10 kgal)
	2C	1,360 kL (359 kgal)
Total tank		1,400 kL (370 kgal)

(a) Sutey MJ. April 8, 1994. "Waste Compatibility Assessment of Tank 241-SY-102 with Tank 241-T-111 via 244-TX-DCRT." Letter 7CF30-94-011 to JH Wicks, Westinghouse Hanford Company, Richland, WA.

Table 2.6. Tank 241-T-111 Best Basis Inventory Source Data

Waste Phase	Waste Type	Associated Volume
Sludge	224-2	904 kL (239 kgal)
	2C	787 kL (208 kgal)
Total tank		1,691 kL (447 kgal)

2.3 241-T-104

Tank T-104 was constructed during 1943 and 1944. It is a 2,006,000-L (530,000-gal) SST in the T tank farm (see Figure 2.1). Tank T-104 was designed for non-boiling waste with a maximum fluid temperature of 104°C (219°F). This tank is the first in the line of cascading tanks (T-104, T-105, and T-106), each 1 ft lower in elevation than the preceding tank. A description of the tank, tank volumes, and sampling events is provided in Table 2.7.

Tank T-104 was used to periodically receive first decontamination cycle (1C) waste and coating removal waste (CW) from the 221-T Bismuth Phosphate Plant from March 11, 1946 through October 19, 1956. No other waste types were received and stored in Tank T-104. During storage in the SSTs, the 1C/CW waste precipitated solids that contained primarily aluminum, bismuth, plutonium, americium, uranium, sodium, phosphate, sulfate, and metals. The 1C/CW supernatant contained primarily aluminum, sodium, nitrate, and cesium-137. As a result, T-104 contained settled 1C/CW solids (i.e., bismuth and plutonium precipitate) and 1C/CW supernatant. The 1C/CW supernatant was removed from Tank T-104 and processed in the 242-T Evaporator (April through July 1951) or disposed in the east section of trench 216-T-14 (January 14, 1954). The interstitial liquid was removed from the 1C/CW sludge present in Tank T-104 and transferred to other underground storage tanks in two campaigns conducted February 1976 to August 1977 and March 24, 1996 to May 30, 1999 (Johnson 2003).

The BBI for Tank T-104 incorporates waste-type templates that correlate with the waste types listed in Table 2.8. Templates are based on sampling data from tanks that contain the same waste type as Tank T-104, supplemented with HDW model data (Agnew et al. 1997a), and contribute significantly to the understanding of the concentrations of certain constituents in the T-104 waste. Tank T-104 contains 1,199 kL (317 kgal) of waste and one defined waste phase—sludge (waste type 1C) according to the current BBI. Other tanks with a 1C inventory include B, BX, C, T, TX, and U farm tanks.

The VSS was used to collect representative samples of the air, gases, and vapors from the headspace of Tank T-104 with sorbent trap and SUMMA™ canisters on February 7, 1996. Sampling devices and controls provided for this job included 11 sorbent trains for selected inorganic analytes (eight sample trains and three field blanks) and five SUMMA canister for permanent gases and total non-methane hydrocarbons (three sample and two ambient canisters).

Two core samples were collected from Tank T-104. Core 45 was obtained from riser 3 on August 20 and 26, 1992, and core 46 was obtained from riser 6 on August 27 and 28, 1992. Core sampling was used because of the phase (solid versus liquid) and depth of the waste and the expectation that a full vertical profile of the waste would be obtained. NPH was used as the hydrostatic head fluid.

Table 2.7. Description of Tank 241-T-104

Tank Description	
Type	Single-shell
Constructed	1943-1944
In-service	3/1946
Diameter	22.9 m (75 ft)
Operating Depth	469.9 cm (185 in)
Design Capacity	2006 kL (530 kgal)
Bottom shape	dish
Ventilation	Passive
Tank Status (as of 1/1/2001)	
Total Waste Volume	1199 kL (317 kgal)
Supernatant Volume	NA
Retained Gas Volume	NA
Retained Gas - Sludge Volume	NA
Retained Gas - Salt Cake Volume	NA
Salt Cake Liquid Volume	NA
Salt Cake Solid Volume	NA
Sludge Volume	1199 kL (317 kgal)
Surface Level (10/3/2000)	311.4 cm (122.6 in.)
PCSACS Surface Level (3/1/2004)	311.4 cm (122.6 in.)
Integrity	Sound
Waste Group Designation	C
Sampling Dates	
Core Samples	8/20/1992–8/26/2002
	8/27/1992–8/28/1992
Vapor Samples	2/7/1996
Service Status	
Declared Inactive	1977
Interim Stabilization	November 1999
Intrusion Prevention	None

Table 2.8. Tank T-104 Best-Basis Inventory Source Data

Waste Phase	Waste Type	Applicable Concentration Data	Associated Volume
Sludge[a]	1C	1992 core composite solids mean concentrations (S/T104/005)	1,199 kL (317 kgal)
		1C sludge template (TS/U204/006)	
		T-104 sludge PCBs (P/T104/005)	
Total Tank			1,199 kL (317 kgal)

(a) The sludge waste phase includes both solids and interstitial liquid; the interstitial liquid volume is estimated to be 176 kL (44 kgal).

3.0 Description of Waste Types

The waste types added to the tanks containing the CH-TRUM waste include lanthanum fluoride finishing waste before and after 1949, and first- and second-cycle waste from the bismuth phosphate process. The waste descriptions and acronyms used in the BBI are provided in Table 3.1.

Table 3.1. BBI Waste Descriptions and Acronyms

Acronym	Waste Type Description
	CH-TRUM
1C	1st cycle decontamination waste from the $BiPO_4$ process
2C1	2nd cycle waste from the $BiPO_4$ process (1944 to 1951)
2C	2nd cycle waste from the $BiPO_4$ process
224-1	LaF_3 finishing waste (pre 1949)
224-2	LaF_3 finishing waste (post 1949)

4.0 Approach

The 11 selected tanks were categorized into three groupings. These groupings and the tanks associated with them are provided in Table 4.1. Existing analytical data from tank waste samples were reviewed and combined with data from similar wastes along with process knowledge of the wastes transferred into the tanks to determine whether selected dangerous waste codes are applicable.

Table 4.1. Tank Groupings

Group	Tanks	Tank Type	Waste Types
B- and T-200 Series Tanks	241-B-201 241-B-202 241-B-203 241-B-204 241-T-201 241-T-202 241-T-203 241-T-204	Single-shell	LaF_3 Finishing (both pre- and post-1948)
T-100 Series Tanks 2C/224	241-T-110 241-T-111	Single-shell	LaF_3 finishing (post-1948) Second-cycle $BiPO_4$
T-100 Series Tanks – 1C	241-T-104	Single-shell	First-cycle $BiPO_4$

The analytical data reviewed included DSC results; concentrations of sulfur, sulfate, cyanide, 2,4,5-trichlorophenol, TOC, and oxalate; the composition of headspace (vapor space); and pH. The dangerous waste codes to which each analysis was applied are provided in Table 4.2. This section discusses the sample analysis methods and the data review approach. Section 5 compares the analytical data with the dangerous waste characteristics.

Table 4.2. Analyses Reviewed in Determining Dangerous Waste Characteristics

Analysis	Waste Code	Characteristic
DSC	D001/D003	Ignitability and Reactivity
Sulfur	D003	Reactivity
Sulfate	D003	Reactivity
Cyanide	D003	Reactivity
2,4,5-trichlorophenol	D041	Toxicity
TOC	D041	Toxicity
Oxalate	D041	Toxicity
Headspace	D001/D003	Ignitability and Reactivity
pH	D002	Corrosivity

4.1 Differential Scanning Calorimetry

DSC results have been used to determine the energetics of the tank wastes as a function of temperature. DSC plots are graphs of the differential heating rate versus temperature. An example scan is provided in Figure 4.1.

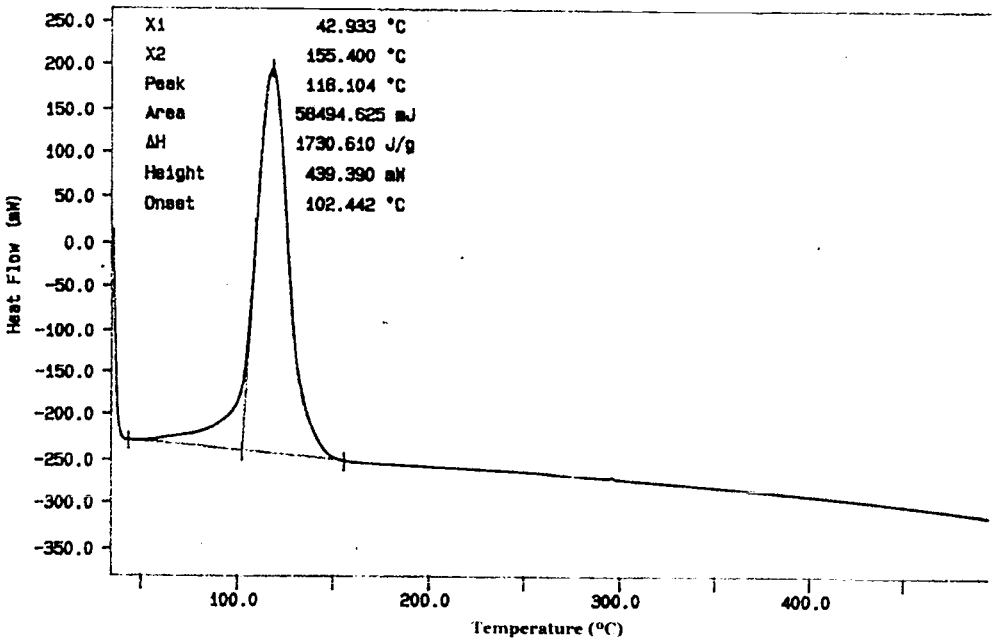

Figure 4.1. DSC of Sample from Lower Half of Segment 8, Core 122 (Tank 241-B-203). Exotherms appear below the baseline and endotherms are above the baseline.

DSC measures the differential power (heat input) necessary to keep a sample and an inert reference substance isothermal as temperature is increased linearly. The data generated by DSC is used to determine heats of reaction, reaction rates, phase transitions, and thermal stabilities. In a DSC scan several transitions may be observed, each indicating at least one separate reaction or phase transition. The area under the peak of each transition is directly proportional to the heat evolved (exothermic) or absorbed (endothermic) by the reaction or phase transition, and the height of the curve is directly proportional to the rate of reaction. In DSC sharp exothermic transitions with large heats of reaction are indicative of unstable materials with vigorous or violent reactions such as explosions or detonations.

The results of the DSC were reviewed for each tank to determine whether any exothermic behavior was observed. If exothermic behavior was observed, the DSC scans were reviewed and the shape and energy of these transitions were determined. If no exothermic behavior was observed or the exothermic transitions were broad with low heats of reactions, the waste was considered stable, with no violent reactions if heated. These results also indicate that the waste is not capable of spontaneous chemical changes that could result in a fire at standard temperature and pressure. If significant endothermic transitions had to occur before the exothermic reaction could occur, the waste could also be considered stable at standard temperature and pressure.

Because all of these tank wastes contain interstitial liquid as aqueous salt solutions, the DSC results obtained on the samples also indicate the reactivity and ignitability of the waste in the presence of water or upon absorption of moisture.

4.2 Sulfur and Sulfate

The major sources of sulfur-containing compounds in Hanford tank wastes are sulfuric acid (H_2SO_4), ferrous sulfamate [$Fe(SO_3NH_2)_2$], sodium sulfate (Na_2SO_4), and sulfamic acid (HSO_3NH_2). Sulfate ions (in the form of sulfuric acid) were used in the bismuth phosphate process to form uranyl sulfate [$UO_2(SO_4)_2^{2-}$] so that it did not coprecipitate with the plutonium. Uranyl sulfate and most of the sulfate ions from the bismuth phosphate process were present in the metal waste stream that was not discharged to any of the SSTs containing the candidate CH TRU mixed waste. Some sulfate ions would have been carried forward with the plutonium in the bismuth phosphate process and been present in the 1C waste stream. Both ferrous sulfamate and sulfamic acid were used to reduce Pu at oxidation states between IV and VI to the Pu(III) oxidation state in the bismuth phosphate process, and sulfate ions were present in both the 1C and 2C wastes streams. Both sulfuric acid (H_2SO_4) and ammonium sulfate [$(NH_4)_2SO_4$] were used in the 231-Z plutonium finishing process, and plutonium-containing waste from this process was recycled to the 224-B and 224-T Concentration Buildings for plutonium recovery. Sulfamic acid was used prior to oxalate precipitation of Pu, and ferrous sulfamate was used in the REDOX and PUREX flow sheets to form the inextractable $Pu(NO_3)_3$ during the solvent extraction process. Sodium sulfate was used in B Plant (1968–1984) to precipitate strontium and separate metal impurities during processes conducted to prepare the strontium for encapsulation. Under tank conditions these sulfur compounds would tend to form sulfates. Tank waste conditions are mildly oxidizing from the abundant nitrate, which also precludes formation of sulfide from sulfate. Therefore, a source of sulfide is not evident in these tanks. If sulfide were present under the pH conditions in the tanks, the predominant form of the sulfide species would be HS^-, and the release of H_2S would be well below toxic concentrations. Detailed information about sulfate chemistry in the Hanford tank wastes is provided in a letter report prepared by Bruce McNamara for CH2M Hill Hanford Group, Inc.[(a)]

Sulfide concentrations in the waste were estimated by the ratio of sulfur to sulfate because no sulfide determination was made on the tank waste samples. Sulfur was determined directly by Inductively Coupled Plasma (ICP) mass spectroscopy on liquid samples from some tanks and on both fusions and acid digestions of solid samples from some tanks. Sulfate was measured by ion chromatography (IC) on liquid samples from the tank as well as water digestions of the solid samples from the tanks. If all of the sulfur were present as sulfate, the mass ratio of sulfate to sulfur would be 3.0 to 1. A mass ratio less than 3 indicates that sulfur containing species other than water soluble sulfate are present in the waste. If the mass ratio is less than 3, the concentration of sulfur containing species in the waste other than water soluble sulfate are calculated from the difference of these two concentrations. The calculated sulfide concentrations are compared with the threshold quantity given in SW-846 (500 mg H_2S/kg of waste formed by acidification), EPA's test method for evaluating solid waste. If the sulfur concentration, after subtracting the sulfate concentration (in moles/kg), is less than (0.0147 moles/kg of waste or 471 mg/kg), the waste cannot be a sulfide-bearing waste. Sulfur and sulfate analyses from the same sample were used

(a) McNamara BK. December 2000. "Evolution of Gases from Cyanide and Sulfide Bearing Waste." Letter Report WTP-RPT-011, Battelle – Pacific Northwest Division, Richland, WA.

for all calculations. Averages were based on the results of these calculations. "Less-than" values were not included in the calculations of the ratios or sulfide estimations.

4.3 Cyanide

Cyanide analyses were performed on a limited number of tank samples. Tanks B-201, B-202, T-104, and T-111 were the only tanks for which cyanide analyses were available. Process histories were analyzed to determine whether cyanide or cyanide-containing compounds may have been added to those tanks for which no analyses are available. If cyanide analyses were available, the measured cyanide concentrations were compared with the threshold values provided in EPA's SW-846 protocol (250 mg HCN/kg of waste). If the measured concentrations were less than 241 mg of CN/kg of waste (241 μg/g of waste or ppm), the waste is not a cyanide-bearing waste.

Sodium or potassium ferrocyanide [$Na_4Fe(CN)_6$ and $K_4Fe(CN)_6$] was added to some of the waste feeds along with $NiSO_4$ and NaOH to precipitate sodium and cesium nickel ferrocyanides [$Na_2NiFe(CN)_6$ and $Cs_2NiFe(CN)_6$], thus scavenging radiocesium from the aqueous wastes. Process histories indicate that none of these tanks accepted cyanide-containing wastes and none of the tanks were part of the cesium precipitation campaigns. None of the selected tanks were on the Ferrocyanide Watch List (Fowler 1993).

4.4 TOC, Oxalate, and 2,4,5-Trichlorophenol

Semivolatile organic analyses (SVOA) were performed on a few of the tank core samples. Concentrations of 2,4,5-trichlorophenol were reported as part of these analyses. If the concentration of the 2,4,5-trichlorophenol was less than the dangerous waste limit (400 mg/L), the waste was not characteristic of toxicity based on 2,4,5-trichlorophenol.

For wastes where SVOAs were not performed, TOC and oxalate analyses were used to estimate the maximum amount of 2,4,5-trichlorophenol that could potentially be available in the waste. The fraction of TOC that was present as oxalate was determined by dividing the concentration of carbon as oxalate by the TOC concentration. The fraction of TOC as oxalate was calculated for each sample that had both oxalate and TOC data. Less-than values for oxalate or TOC were not used for this estimation. This fraction was used to determine the concentration of carbon present as other organic compounds. For determining whether these wastes should carry dangerous waste designation D041 toxicity characteristics based on 2,4,5-trichlorophenol concentration, all of the organic compounds other than oxalate were considered to be 2,4,5-trichlorophenol. This is a very conservative estimate. The potential concentration of 2,4,5-trichlorophenol was then calculated according equation 4.1:

$$\text{Concentration} = (\text{TOC in } \mu g/g) * (1 - \chi_{oxalate}) * (\rho_{waste}) * (2.74 \text{ g tcp/g C}) \qquad (4.1)$$

where

Concentration	= the potential concentration of 2,4,5-trichlorophenol in μg/ml or mg/L
TOC	= the concentration of total organic carbon in the waste
$\chi_{oxalate}$	= the fraction of the total organic carbon that is present as oxalate
ρ_{waste}	= the bulk density of the waste in g/mL
2.74 g tcp/g C	= the mass of 2,4,5-trichlorophenol per mass of TOC.

4.5 Headspace

Analysis of the gases present in the headspace was available on a few of the tanks of interest. Headspace samples were taken using sorbent traps for inorganic analytes and SUMMA canisters for permanent gases and total non-methane organic compounds. The canister samples were analyzed by gas chromatography, and the sorbent trap samples were determined by desorbing the analytes of interest from the trap with the appropriate aqueous solution and analyzing the solution by ion chromatography or ion selective electrode. The concentrations of these analytes were compared with their threshold limit value-time weighed average (TLV-TWA) as reported by the American Conference of Governmental Industrial Hygienists in the *2004 Threshold Limit Values for Chemical Substances in the Work Environment* (ACGIH 2004). Threshold limit values for selected analytes are provided in Table 4.3.

Table 4.3. Threshold Limit Values for Selected Headspace Gases

Analyte	TLV-TWA (ppmv)
NH_3	25
NO_2	3
NO	25
CO_2	5000
CO	25
CH_4	1000
H_2	Simple asphyxiant
N_2O	50

Process wastes may be diluted prior to being retrieved from the tanks. Henry's Law states that for solutes with measurable vapor pressures, the vapor pressure of the solute in dilute solutions is proportional to the mole fraction of that solute. Based on this law used for ideal gases in dilute solutions, the concentration of gases in the headspace of the waste container will be proportional to the dilution; therefore, for process wastes that are diluted prior to retrieval from the tank, the concentration of the headspace gases can be divided by the dilution factor. If the concentrations of the headspace gases are less than the TLV-TWA for those gases, the waste does not generate toxic gases, vapors, or fumes in a large enough quantity to present a danger to human health or the environment.

4.6 pH

Measurements of the pH of the tank wastes were made on most of the tanks discussed in this report. Many of these pH measurements were made on liquid grab samples. These pH measurements provide an accurate estimate of the pH of the waste in the tank. For some tanks, pH was measured on a water digest of samples that were diluted and vacuum dried similar to the proposed process for the TRUM tank wastes. This measurement followed a protocol similar protocol to that of SW-846 method 9045 in "Test Methods for Evaluating Solid Waste, Physical/Chemical Methods." If the pH of the tank waste samples is between 2 and 12.5, the waste is not corrosive.

5.0 Summary of Characterization Data

The existing analytical results from sampling events were reviewed and compared with the data on waste transfers in these tanks. These results are summarized for each of the three tank categories. The majority of the data were extracted from the TWINS database and characterization reports prepared by PNNL.

The analyses reported in this document were performed on wet slurries and sludges unless otherwise noted. Some of these results are reported on a dry weight basis; therefore, these results are independent of the water content of the samples. Most of the results were reported as concentration of analyte per gram of sample (wet). Once the sample is dried these concentrations will increase by the ratio of wet sample mass to dry sample mass (the reciprocal of the weight percent total solids of the sample).

Vacuum drying the retrieved tank wastes is the method proposed for removing water from the waste prior to packaging and shipping. In this process, the majority of the free water will be removed from the sample, and some volatile constituents may evaporate; but the ratios of the non-volatile constituents (e.g., sulfur and sulfate) will not change. The ratio of oxalate to TOC may actually increase if volatile organic carbon is present in the sludge, but the ratio will not decrease due to the vacuum drying process.

During the DSC analysis, water evaporates from the sample as the temperature increases; therefore, after the first endothermic transition (water loss), the results obtained from the DSC provide the energetics of the dried sample. Corrosivity of dried samples is obtained by adding water to the sample and measuring the pH of the resulting supernatant liquor; therefore, the pH of the wet sludge provides the data required to determine the corrosivity of the dried solids.

5.1 200 Series Tanks (B and T Tank Farms)

Thermal analysis was performed in duplicate on direct subsamples from segments and core composites of all of the 200 series tanks in the B and T farms and on some liquid grab samples from the tanks. TGA and either DSC or differential thermal analysis (DTA), which provide similar data, were performed on each sample. A single endothermic transition was observed in many of the samples, but two were observed in some of them. In one sample that was dried (segment 3 of core 27 from Tank B-201), a third endothermic transition was observed. The first endotherm was observed in all the samples except the dried sample and occurred between ambient temperature and 150°C. This endothermic transition coincides with mass loss observed in the TGA and indicates the evaporation of free water from the sample. The enthalpy of this endothermic transition was dependent on the water content of the sample. The second endothermic transition was observed between 200° and 330°C. This endotherm is also associated with weight loss on the TGA; therefore, it is probably due to loss of more tightly bound water such as waters of hydration. The third endothermic transition was much smaller and was observed at higher temperatures. A small weight loss (~2 wt%) is associated with this transition. Because this same weight loss was observed in many of the other samples, this endotherm may be present in other samples but not readily observed because the enthalpy of this endotherm is so much smaller than the water loss endotherms in wet samples. These transitions are probably due to the loss of waters of hydration, phase transitions, or decomposition of salts and hydroxides to oxides.

No significant exothermic behavior was observed in the 200 series tanks in T farm or in Tanks B-201 and B-204. The maximum enthalpy observed for these samples was 16.7 J/g of waste on a dry weight basis in an aliquot from segment 9 of core 190 (Tank T-203). An exotherm was observed in duplicate aliquots of several segments from Tank B-202 (segments 2, 4, 5, and 8 of core 24 and segments 1, 2, 3, 6, 7, and 8 of core 25) and in two of 48 samples from B-203. All of the exothermic transitions were broader peaks with smaller amplitudes than the endothermic transitions. The exothermic transitions from Tank B-202 and the solid samples from B-203 began near 250°C and ended near 450°C with enthalpies between -30 and -172.7 J/g of waste. A liquid grab sample analyzed by DSC also appeared to exhibit significant exothermic behavior (ΔH = -176 J/g). The duplicate and triplicate samples did not exhibit the same behavior. The duplicate sample indicated an endotherm at this temperature range; the triplicate sample indicated a slight exotherm between 430° and 500°C, but the enthalpy was only -33.7 J/g.

Sulfur and/or sulfate analyses were performed on segments from the core samples divided into the lower and upper half, core composites, and/or drainable liquids from the core samples. Analyses were performed on some samples from all of the B- and T-200 series tanks. The type and number of samples analyzed for sulfur and sulfate from each tanks are reported in Table 5.1. Many of the results from these samples were below the quantification limits, and only less-than values were reported. Tanks for which only less-than values were reported are noted in the table.

Table 5.1. Sulfur and Sulfate Analyses for the B- and T-200 Series Tanks

Tank	Number of Samples					
	Sulfur Analysis			Sulfate Analysis		
	Segment	Composite	Liquid	Segment	Composite	Liquid
B-201	None	None	None	None	4	None
B-202	None	6	None	None	6	None
B-203	41	None	3	41	None	3
B-204	49	None	3	49	None	3
T-201	6[(a)]	None	5	6	None	5[(a)]
T-202	6	None	None	6	None	None
T-203	20	None	1	20	None	1
T-204	None	1[(a)]	None	None	1[(a)]	None

(a) All results were reported as less than values.

The majority of the results indicate that the sulfur in these wastes exists primarily as sulfate. Only seven samples had SO_4^2-to-S ratios less than 2.8, and many of these had ratios greater than 2.4. These ratios indicate that the sulfur is present predominantly in the sulfate form.

Cyanide analyses were performed on core composites and drainable liquids from Tanks B-201 and B-202 and on a single segment from Tank B-203. The results of these analyses are reported in Table 5.2. Cyanide analyses were not performed on Tank B-204 or any of the T-200 series tanks. These analyses were done in duplicate by three different methods. Water digests of the samples from Tanks B-201 and B-202 were analyzed by ion chromatography and spectrophotometry at 580 nm, respectively. The solids from the single segment from Tank B-203 (core 122, lower half of segment 10) were analyzed by a method that incorporates cyanide complexes. The majority of the results were below the quantification

Table 5.2. Cyanide Analyses for the B- and T-200 Series Tanks

Tank	Core	Sample	Concentration (µg/g)
B-201	26	Composite 1	5
		Composite 2	3.95
	27	Composite 1	< 3
		Composite 2	< 2
B-202	4	Composite	< 0.5
	5	Composite	< 2.5
		Composite	< 2.44
	10	Composite	< 0.5
	11	Composite	< 8
		Composite	< 2.38
	16	Drainable liquid	<0.5 µg/ml
B-203	122	Segment 10	< 5.91

limit, and less-than values were reported. The less-than values ranged from 0.5 to 8 µg/g, with most less than 2.5 µg/g. Only two results were reported as actual values. Both of these results were from Tank B-201, core 26. These results (4 and 5 µg/g) were within the range of the less-than values reported and well below the 241 µg/g required for a waste to be considered a cyanide-bearing waste. None of the B- or T-200 series tanks were on the Ferrocyanide Watch List (Fowler 1993), and cyanide-containing wastes and/or process streams were not introduced into these tanks.

Semi-volatile organic analysis (SVOA) was performed on samples from Tanks B-201 and B-202. The concentrations of 2,4,5-trichlorophenol in these samples were less than 240 and 150 µg/g, respectively. No other SVOA results were reported on the B- and T-200 series tanks. Both TOC and/or oxalate analyses were performed on the other tanks. The range of concentrations obtained for these analyses are reported in Table 5.3. No TOC analyses were performed on samples from Tanks B-204 or

Table 5.3. TOC and Oxalate Results for the B- and T-200 Series Tanks

Tank	Sample Type	TOC (µg/g)		Oxalate (µg/g)	
		Persulfate Oxidation	Furnace Oxidation	As Oxalate	As Carbon
B-201	Core Composite	575 – 4,500	-	-	-
B-202	Core Composite	-	2,020 – 3,770	-	-
B-203	Segment	72	-	811 - 16,600	157 - 3200
B-203	Drainable Liquid	96 µg/mL	-	486 µg/mL	94 µg/mL
B-204	Segment	-	-	450 - 2,400	87 - 463
B-204	Drainable Liquid	-	-	753	145
T-201	Composite & Segments	133 - 304	-	682 – 1,140	132 - 220
T-201	Drainable Liquid	-	-	< 223	< 43
T-202	Composite & Segment	347	-	372 – 1,510	72 - 291
T-203	Composite & Segment	421	-	283 – 1,590	55 - 307
T-203	Drainable Liquid	-	-	< 223	< 43
T-204	Composite	312	-	1,330	257

on the drainable liquid composites from Tanks T-201 and T-203. Only results below the quantification limit were obtained for oxalate in the drainable liquids from Tanks T-201 and T-203, and no oxalate analyses were performed on Tanks B-201 and B-202. The ranges reported in Table 5.3 include results above the quantification limit unless only less-than values were reported.

The fraction of the TOC that was present as oxalate could not be determined for the solids from the B-200 series tanks because the quantifiable oxalate and TOC values were not obtained on the same samples. In the T-200 series tanks and the drainable liquid from Tank B-203, the majority of the TOC appears to be present as oxalate. For the samples that had both TOC and oxalate values, the fraction of TOC that was present as oxalate was approximately 75%.

Mercury concentrations in the composites of cores 26 and 27 from Tank B-201 and 4, 5, 10, and 11 from Tank B-202 were measured by cold vapor atomic absorption (CVAA). Leachable mercury from TCLP solutions that were digested in acid from core 27 composites was also measured by CVAA. Leachable mercury from TCLP and acid digestion from core 5 and 10 composites were measured by atomic absorption (AA) and acid digestion of core 4, 5, and 10 composites. The direct samples are a better indicator of mercury concentration, but TCLP samples provide data on leachable mercury. Mercury results are reported in Table 5.4. All TCLP samples were below 0.077 μg/mL. The direct samples and acid digestions of the core composites ranged from <0.123 to 1.1 μg/g. Mercury concentrations were significantly higher in B-201 than B-202. No mercury data are available on the other tanks.

Table 5.4. Mercury Analyses for the B- and T-200 Series Tanks

Cold Vapor Atomic Absorption			
Tank	**Core**	**Sample**	**Concentration (μg/g)**
B-201	26	Composite 1	1
		Composite 2	0.75
	27	Composite 1	0.54
		Composite 2	0.11
		TCLP Composite 1	0.025 μg/mL
		TCLP Composite 2	0.003 μg/mL
B-202	4	Composite	0.27
	5	Composite	0.33
		Composite	0.74
	10	Composite	<0.12
		Composite	0.26
	11	Composite	0.26
Atomic Absorption			
B-202	4	Composite	<0.49
	5	Composite	<0.49
		TCLP Composite	0.077
		TCLP Composite	0.056
	10	Composite	<0.49
		TCLP Composite	<0.05

The pH of the waste was measured on core or tank composites from Tanks B-201, B-202, B-203, T-203, and T-204. The pH of water leaches (ratio of 100:1 water to sample) was also measured on B-201. The results of these pH measurements are provided in Table 5.5. The pH of the supernatant liquor from multiple dilutions of a Tanks B-203, T-203, and T-204 composite were also measured and did not vary over the range of dilutions (as-received waste to a 4:1 water-to-waste dilution). The average pH value for each tank composite measured in this supernatant liquor is also reported in Table 5.5 (Tingey et al. 2003). The pH of these samples did not exceed the 12.5 limit for corrosive wastes.

Gas analysis was performed on vapor taken from headspace of Tanks B-202, B-204, T-201, T-202, T-203, and T-204. The highest measured concentration of each of the gases in any of the B- or T-200 series tanks is reported in Table 5.6. These gases represent less than 2 % of the lower flammability limit (LFL). None exceeded the TLV-TWA.

Table 5.5. pH of the B- and T-200 Series Tanks

Tank	pH		
	Composite	**Water Leach**	**Dilutions**
B-201	12.3	8.5 ± 0.05	--
B-202	12.3 ± 0.15	--	--
B-203	10.5	--	11.4 ± 0.10
T-203	10.9	--	11.0 ± 0.06
T-204	9.84	--	11.0 ± 0.05

Table 5.6. Headspace Analysis for the B- and T-200 Series Tanks

Analyte	Vapor Concentration (ppmv)
NH_3	8
NO_2	< 0.16
NO	< 0.16
H_2O	14 mg/L
CO_2	696
CO	< 3
CH_4	< 4
H_2	< 17
N_20	< 17
Total non-methane organic compounds	0.86 mg/m^3

Because exothermic behavior was observed only at temperatures exceeding 250°C, and all of the exothermic transitions were broader peaks and smaller amplitudes than the endothermic transitions, the waste can be considered to be thermally stable. These results also indicate that the waste will not form potentially explosive mixtures with water, will not detonate or undergo an explosive reaction if heated, and will not detonate or decompose explosively at standard temperature and pressure. These results, along with the low concentrations of flammable gases present in the headspace (<2% of the LFL),

indicate that this waste does have ignitable characteristics. The waste is not cyanide- or sulfide-bearing and does not contain 2,4,5-trichlorophenol at concentrations exceeding the dangerous waste limits. None of the toxic gases observed in the headspace are above the TLV-TWA; therefore, toxic gases, vapors, or fumes should not be generated at concentrations hazardous to human health.

5.2 T-100 Series Tanks (241-T-110 and 241-T-111)

Thermal analysis was performed in duplicate on samples from both the upper and lower halves of all segments from cores 180 and 181 of Tank T-110. Thermal analysis was also performed on samples from all segments of cores 31 and 33 of Tank T-111. The DSC analysis from segment 2 of core 33 was repeated on dried samples multiple times. Analysis was also repeated on dried samples from segment 1 of core 33. Both TGA and DSC were performed on each sample.

A single endothermic transition was observed in many of the T-110 samples, but two were observed in some of them. The first endotherm was observed in all of the samples and occurred between ambient temperature and 150°C. This endothermic transition coincides with mass loss observed in the TGA and is indicative of the evaporation of the free water from the sample. The enthalpy of this endothermic transition was dependent upon the water content of the sample. The second endothermic transition was observed between 240° and 340°C with an onset temperature of approximately 300°C. No apparent mass loss was observed in the TGA over this temperature range; therefore, this second endotherm is probably due to a phase transition or the decomposition of hydroxides to oxides. An exotherm was observed in the sample from the upper half of segment 5 of core 181. The enthalpy of this exotherm was 12.6 J/g over a temperature range from 450° to 500°C. This exotherm was not observed in the duplicate sample.

A single endothermic transition followed by a significant but broad exothermic reaction was observed on segments 1, 2, and 3 of core 31 and segments 1 and 2 of core 33 from Tank T-111. The lower segments (segment 4 of core 31 and segments 3 and 4 of core 33) exhibited minimal or no exothermic behavior. The initial endotherm was similar to the first one observed in Tank T-110 and indicates evaporation of free water. The exotherm does not coincide with a mass loss and appears to represent the oxidation of the organic carbon in the waste. The enthalpy of this exothermic reaction on wet samples was significantly smaller than the water endotherm and ranged from 250 to 650 J/g. Segment 2 of core 33 had the largest enthalpy. The water loss endotherm generally exceeded 1000 J/g. The temperature range of the exotherm was consistent for all of the samples and ranged from 200° to 400°C.

Similar results were observed for the dried samples from segment 2 of core 33 except the water loss endotherm was much smaller because most of the water was removed before the analyses. On a few samples, all of the water was removed prior to the measurements, and this water loss endotherm did not exist. On these samples the onset temperature of the exotherm was 199°C with an enthalpy of approximately 900 J/g.

Neither sulfur nor sulfate analyses were performed on core samples from Tank T-110 but were obtained on a grab sample. Both sulfate and sulfur data are available on the core composites from Tank T-111 (cores 31 and 33) as are sulfur analysis data on segment 9 of core 31. Results from these analyses as well as the ratio of sulfate to sulfur are reported in Table 5.7. Based on these analyses, the majority of the sulfur is present as sulfate.

Table 5.7. Sulfur and Sulfate Concentrations in Tanks T-110 and T-111

Tank	Sample	Sulfur (μg/g)	Sulfate (μg/g)	Sulfate/Sulfur Ratio
T-110	Grab Sample 10T-96-1	1590	4360	2.7
T-111	Core 31 Composite 1	1230	3690	3.0
	Core 31 Composite 2	1270	3740	2.9
	Core 31 Segment 9	1480	5120	3.5
	Core 33 Composite 1	1140	3290	2.9
	Core 33 Composite 2	1220	3470	2.8

Cyanide analyses were performed on composites from cores 31 and 33 of Tank T-111 and on water digests of segments 1 and 2 from core 33. The cyanide analyses were performed by two methods on the water digest from segment 2. Cyanide analyses were not performed on Tank T-110 samples. The majority of analyses were done by spectrophotometry at 580 nm, but one of the water digests on segment 2 was analyzed using a method incorporating cyanide complexes. The majority of the results (Table 5.8) are below the detection limit and were reported as less-than values. These less-than values ranged from 0.049 to 5 μg/g. Only three results were reported as actual values. Two of these results from the core 33 composites were within the range of the less-than values (3.72 and 2.47 μg/g); the third was obtained from segment 1 and was much higher than any of the other results (57.6 μg/g) but still well below the 241 μg/g required for a waste to be considered cyanide bearing. Neither of these tanks is on the Ferrocyanide Watch List (Fowler 1993), and cyanide-containing wastes, and/or process streams were not introduced into these tanks.

Table 5.8. Cyanide Analyses of Tank T-111 Samples

Core	Sample	Concentration (μg/g)
31	Composite 1	<4.3
	Composite 2	<3.5
33	Composite 1	<4.9
	Composite 2	<4.7
	Segment 1	3.72
	Segment 2	2.47
	Segment 1 Water Digest	57.6
	Segment 1 Water Digest	<2.7
	Segment 2 Water Digest	<0.05

SVOA was performed on composite samples from cores 31 and 33 of Tank T-111. The concentrations of 2,4,5-trichlorophenol in these samples were below the detection limit, and less-than values ranged from 65 to 120 μg/g. No SVOA data were available for Tank T-110. These values are well below the dangerous waste limit of 400 μg/g.

No oxalate data were obtained on Tank T-111, and a less-than value for oxalate (< 556 μg/mL) and a single value for TOC from one grab sample (10T-96-1) were the only data available on Tank T-110. The TOC for Tank T-110 was reported to be 45 μg/mL. Both the 222-S Laboratory and PNNL performed

TOC analyses on samples from cores 31 and 33 from Tank T-111. Furnace oxidation of water digest samples was performed at the 222-S Laboratory. Both hot persulfate and furnace oxidation methods were performed at PNNL on dried samples from segment 2 of core 33.[a] The furnace oxidation method on dried samples appeared to provide the most accurate values that were consistent with the exothermic behavior described earlier. The TOC results on the water leach on cores 31 and 33 were 3740 and 2500 μg/g, respectively. TOC results of the hot persulfate oxidation of the segment 2 sample were consistent with these results (3460 μg/g). TOC of the grab sample was much lower (420 μg/g). The TOC of the wet sludge (18% solids) measured by furnace oxidation at PNNL was 0.89%. The dried sludge results were approximately four times lower by hot persulfate than by furnace oxidation, which is consistent with the wet sludge results (4.09% TOC, which, when corrected for water content, is equivalent to 0.74% for the wet sludge). The TOC is expected to be higher in segment 2 than in the core composites based on the exothermic behavior of segment 2 compared with the lower segments from the core. The exothermic behavior of the top three segments was much higher than that observed in the lower three segments.

Mercury concentrations in the composites of cores 31 and 33 from Tank T-111 were measured directly by CVAA and on acid digestions of the TCLP leach samples by AA. The direct samples give a better indication of the mercury concentrations in the sample, but the TCLP samples provide data on the leachable mercury in the waste. All of the TCLP samples were below the detection limit of approximately 0.045 μg/mL. The direct samples ranged from 1.08 to 1.59 μg/g. No mercury data are available on Tank T-110.

The pH of the waste was measured on water digests of the core 31 and 33 composites from Tank T-111 and on grab samples from both T-110 and T-111. The measured pH values were consistent for both composites from T-111. The measured pH was 9.98 ± 0.17. The pH of the grab sample taken in 1994 (11.7) was significantly higher than the water digest of the core composites, but both were well below the 12.5 limit for corrosive wastes. The pH measured on the grab samples from Tank T-110 (samples 10T-96-1, 10T-96-2, and 10T-96-3) were significantly lower than the pH of Tank T-111 but consistent with one another. The measured pH for T-110 was 8.42 ± 0.03. The pH of the supernatant liquor from multiple dilutions of a Tank T-110 composite was found to be consistent with the grab sample results and did not vary over the range of dilutions (as-received waste to a 4:1 water-to-waste dilution). The pH values measured in these supernatant liquors was 8.28 ± 0.02 (Tingey et al. 2003). Gas analysis was performed on vapor taken from headspace of both T-110 and T-111 (Huckaby and Bratzel 1995). The results of these analyses are summarized in Table 5.9. The gases represent less than 0.2% of the LFL.

(a) Baldwin DL. January 14, 1994. "Final T-111 (Core 33, Segment 2) Dry/As-Received TOC Results from Two Methods," Revision 2. Letter to RM Bean.

Table 5.9. Headspace Analysis for Tanks T-110 and T-111

Analyte	Vapor Concentration (ppmv)	
	T-110	**T-111**
NH_3	108	226
NO_2	< 0.05	< 0.09
NO	< 0.06	< 0.06
H_2O	16.9 ± 0.2 mg/L	11.8 mg/L
CO_2	358 ± 3	69
CO	< 25	< 12
CH_4	< 25	--
H_2	< 25	< 94
N_2O	< 25	< 12.6
Total non-methane organic compounds	1.12 ± 0.27 mg/m^3	--

The retrieval process planned for Tanks T-110 and T-111 will likely result in dilution of the waste. Based on Henry's Law for dilute solutions, concentrations of an ideal gas will be diluted proportional to the dilution of the solute; therefore, an estimate of the concentration of these headspace gases above the process waste retrieved from the tank should include this dilution factor. A dilution factor of 4.3 would be required to dilute the NH_3 concentration in the vapor below the TLV-TWA of 25 ppmv in Tank T-110, and a dilution factor of about 9 in Tank T-111 would be required to achieve the same effect. All of the other gases are below the TLV-TWA.

Because no exothermic behavior was observed at normal temperatures and pressures and the quantity of flammable gases present in the headspace is less than 0.2% of the LFL, this waste is not ignitable. The energetics of the system also indicates that the waste is thermally stable below 200°C, does not form potentially explosive mixtures with water, and does not detonate or decompose explosively at standard temperature and pressure. The waste is not a cyanide- or sulfide-bearing waste and does not contain 2,4,5-trichlorophenol at concentrations exceeding the dangerous waste limits. The only toxic gas observed in the headspace above the TLV-TWA is NH_3. The process wastes retrieved from the tank will require a dilution factor of at least 4.3 for Tank T-110 and 9 for Tank T-111 (or other mitigation methods) to ensure that toxic gases, vapors, or fumes will not be generated at concentrations that are hazardous to human health.

The energetics of Tank T-110 waste is significantly different than T-111 waste, and T-110 waste is thermally stable over the entire temperature range of the DSC measurements (ambient to 500°C). Oxidation of organic compounds in T-111 waste results in a significant release of energy at temperatures above 200°C when the waste is dried. Under current tank conditions (significant moisture associated with the sludge), the endothermic transition due to the evaporation of water is much larger than the exothermic reaction. The exothermic behavior of Tank T-111 wastes is not observed at temperatures below 200°C, does not form potentially explosive mixtures with water, and does not detonate or decompose explosively at standard temperature and pressure.

5.3 T-100 Series Tank 1C (241-T-104)

Thermal analysis was performed in duplicate on samples of both segments and composites from cores 45 and 46. Both TGA and DSC were performed on each sample. A single endothermic transition was observed for all samples; this endothermic transition coincides with mass loss observed in the TGA and is indicative of the evaporation of free water from the sample. The enthalpy of this endothermic transition was dependent upon the water content of the sample, and the temperature range of the transition was ambient to 150°C. No exothermic behavior was observed in any of these samples. Detailed results, including the plots of the DSC and TGA analyses, can be found in WHC (1993).

Sulfur analyses were performed on acid digests of the core composite samples from cores 45 and 46 and on water digests of these same samples. Both sulfur and sulfate analyses were performed on the drainable liquid composite from core 46. The sulfate/sulfur ratio was calculated to determine whether sulfur was present in a form other than sulfate. The ratios indicated the sulfur was present predominantly in the sulfate form. These results are reported in Table 5.10.

Cyanide analyses were performed in duplicate on core composites from cores 45 and 46 and on a drainable liquid composite from core 46. The majority of the results were below the detection limit, and a less-than value of 2 μg/g was reported for all core composite samples (Table 5.11). The cyanide concentration in the drainable liquid sample was 0.7 μg/mL. These results are well below the 241 μg/g required for a waste to be considered cyanide bearing. Tank T-104 is not included on the Ferrocyanide Watch List (Fowler 1993) and does not contain cyanide wastes or process stream waste.

Table 5.10. Sulfur and Sulfate Concentrations in Tank T-104

Sample	Sulfur (μg/g)	Sulfate (μg/g)	Sulfate/Sulfur Ratio
Core 45 Composite 1	1250	3880	3.1
Core 45 Composite 2	1350	3920	2.9
Core 46 Composite 1	1270	3920	3.1
Core 46 Composite 2	1250	3870	3.1
Core 46 Drainable Liquid	1700	4260	2.5

Table 5.11. Cyanide Analyses of Tank T-104 Samples

Core	Sample	Concentration (μg/g)
45	Composite 1	< 2.0
	Composite 2	< 2.0
46	Composite 1	< 2.0
	Composite 2	< 2.0
	Drainable Liquids	0.70 μg/mL

SVOA was performed on composite samples from cores 45 and 46. The concentration of 2,4,5-trichlorophenol in the samples was less than 53 μg/g. TOC was analyzed by furnace oxidation on water digests of the core 45 and 46 composites. These analyses were performed on water digestion, so only the water-soluble fraction of the organic carbon would be present, but the solubility of 2,4,5-trichlorophenol

in water was 1,000 μg/g at 25°C. The majority of the results were below the detection limit, 550 μg/g. The duplicate run of the core 45 composite did give a quantifiable result of 706 μg/g. A TOC concentration of 451 μg/mL was measured on an undigested sample of the drainable liquid composite from core 46. No oxalate analyses were performed. The 2,4,5-trichlorophenol concentrations measured are well below the dangerous waste limit (400 μg/g). The results indicate that the water-soluble TOC concentration in this tank is just slightly higher than the limit for 2,4,5-trichlorophenol; therefore, over 55% of the water soluble organic carbon would need to be 2,4,5-trichlorphenol to exceed this limit.

Mercury concentrations in the composites of core 45 and 46 were measured by CVAA. The majority of the analyses were below the detection limit of 0.125 μg/g. The duplicate analysis from the core 45 composite 1 was slightly above the detection limit, with a reported value of 0.127 μg/g.

The pH of Tank T-104 was measured directly on undigested composites from cores 45 and 46 and on the composite of drainable liquids from core 46. The measured pH value was 9.99 ± 0.03. The pH of the drainable liquids (9.95) appeared to be slightly lower than the core composites (10.00 ± 0.01) but well within normal measurement error. Table 5.12 lists results. Gas analysis was performed on vapor from the headspace of the tank (Pool et al. 1996). The results are summarized in Table 5.13. These gases represent less than 0.2 % of the LFL.

Table 5.12. pH of Tank T-104 Samples

Core	Sample	pH
45	Composite 1	10.00
	Composite 2	10.00
46	Composite 1	10.02
	Composite 2	9.99
	Drainable Liquids	9.95

Table 5.13. Headspace Analysis for Tank T-104

Analyte	Vapor Concentration (ppmv)
NH_3	105 ± 3
NO_2	0.3 ± 0.1
NO	0.3 ± 0.2
H_2O	12.0 ± 0.5 mg/L
CO_2	241
CO	< 3
CH_4	< 4
H_2	12*
N_20	8*
Total non-methane organic compounds	1.93 mg/m^3
(a) Target compound detected above the instrument detection limit (IDL) but below the estimated quantitation limit (EQL).	

The retrieval process planned for Tank T-104 will likely result in dilution of the waste. Based on Henry's Law for dilute solutions, concentrations of an ideal gas will be diluted proportional to the dilution of the solute; therefore, an estimate of the concentration of these headspace gases above the process waste retrieved from the tank should include this dilution factor. A dilution factor of 4.2 would be required to dilute the NH_3 concentration in the vapor below the TLV-TWA of 25 ppmv.

Because no exothermic behavior was observed and the quantity of flammable gases present in the headspace is less than 0.2% of the LFL, this waste is not ignitable. The energetics of the system also indicates that the waste is thermally stable, does not form potentially explosive mixtures with water, does not detonate or undergo an explosive reaction if heated, and does not detonate or decompose explosively at standard temperature and pressure. The waste is not cyanide- or sulfide-bearing and does not contain 2,4,5-trichlorophenol at concentrations exceeding the dangerous waste limits. The only toxic gas observed in the headspace above the TLV-TWA is NH_3. The process wastes retrieved from the tank will require a dilution factor of at least 4.2 (or other mitigation method) to ensure that toxic gases, vapors, or fumes will not be generated at concentrations that are hazardous to human health.

6.0 Conclusions

No exothermic behavior was observed in the majority of the CH-TRUM wastes. Exothermic transitions were observed in a limited number of samples from Tank B-203, and a significant number of samples showed consistent exothermic behavior in Tanks B-202 and T-111. With the exception of the dried samples from Tank T-111, a large endothermic transition was observed between ambient temperature and approximately 150°C. This transition corresponded to a large mass loss in the TGA, indicating the loss of free water from the sample. This is the major transition in all of the wet sludge samples. A second endothermic transition was observed in many of the samples. This endothermic transition began at about 200°C and was due to phase transitions, decomposition of salts or hydroxides, or loss of more tightly bound water. The exothermic transitions that were observed in these samples were generally broad peaks with small amplitudes at temperatures exceeding 200°C, indicating that explosive reactions or ignition hazards at standard temperatures or pressures are unlikely. The waste with the greatest potential for exothermic reactions is Tank T-111. These reactions would most likely be due to the oxidation of organic compounds that are found primarily in the top section of the waste. One potential source of the organic is the NPH used during core drilling. NPH would float to the top of the waste due to its low density, consistent with the results obtained for this core. One probable reaction is the oxidation of the NPH with the nitrate present in the tank. If these wastes are wet, significant energy must be expended to remove the water from the waste before any reaction can occur.

No data were available on the concentrations of sulfide in these tank wastes, but the sulfur and sulfate analyses indicated that the majority of the sulfur in the waste is present as sulfate. Based on the pH, temperature, and mildly oxidizing conditions of the tank wastes, sulfate is stable and will not react to form sulfide; therefore, the CH-TRUM wastes are not sulfide-bearing.

Limited cyanide analyses were available for these tanks, but the results that were available indicated that these wastes do not contain cyanide at concentrations sufficient to be considered cyanide-bearing wastes. None of these tanks are included on the Ferrocyanide Watch List (Fowler 1993) and cyanide containing wastes and/or process streams were not introduced into any of these tanks.

Concentrations of the 2,4,5-trichlorophenol measured in the CH-TRUM wastes did not exceed the dangerous waste limits; therefore, these wastes do not meet the criteria of a toxic characteristic based on this organic constituent. Oxalate and TOC data indicate that the majority of the organic carbon in the waste exists as oxalate. Based on the existing analytical data obtained from the TWINS database and PNNL and Hanford reports, the CH-TRUM does not exceed the pH criteria of a corrosive waste.

Gas analysis of the headspace vapors in these tanks and sniff data taken prior to sampling events indicate that all of toxic vapors except NH_3 are well below the TLV-TWA. Ammonia is produced in Hanford wastes by several pathways and is retained in the waste due to its solubility in water. The major pathways for ammonia generation include thermal and radiolytic reactions involving nitrite ions and nitrogen-containing complexants (Stock and Pederson 1997). Dilution of the tank wastes prior to retrieval or other mitigation methods will be required to meet this criterion on all of the wastes except those in the B- and T-200 series tanks.

Analysis of the existing characterization data from core, grab, and vapor samples from the CH-TRUM wastes supports the removal of the dangerous waste codes for ignitability (D001), corrosivity (D002), and toxicity arising from the presence of 2,4,5-trichlorophenol (D041). For the B- and T-200 series tanks, the data also supports the removal of the dangerous waste codes for reactivity (D003). All constituents and properties were below the levels required to remove waste code D003 from Tanks T-104, T-110, and T-111 except ammonia, which exceeds the TLV in the headspaces of these tanks.

7.0 References

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Appendix

Core Sample Profiles

Appendix

Core Sample Profiles

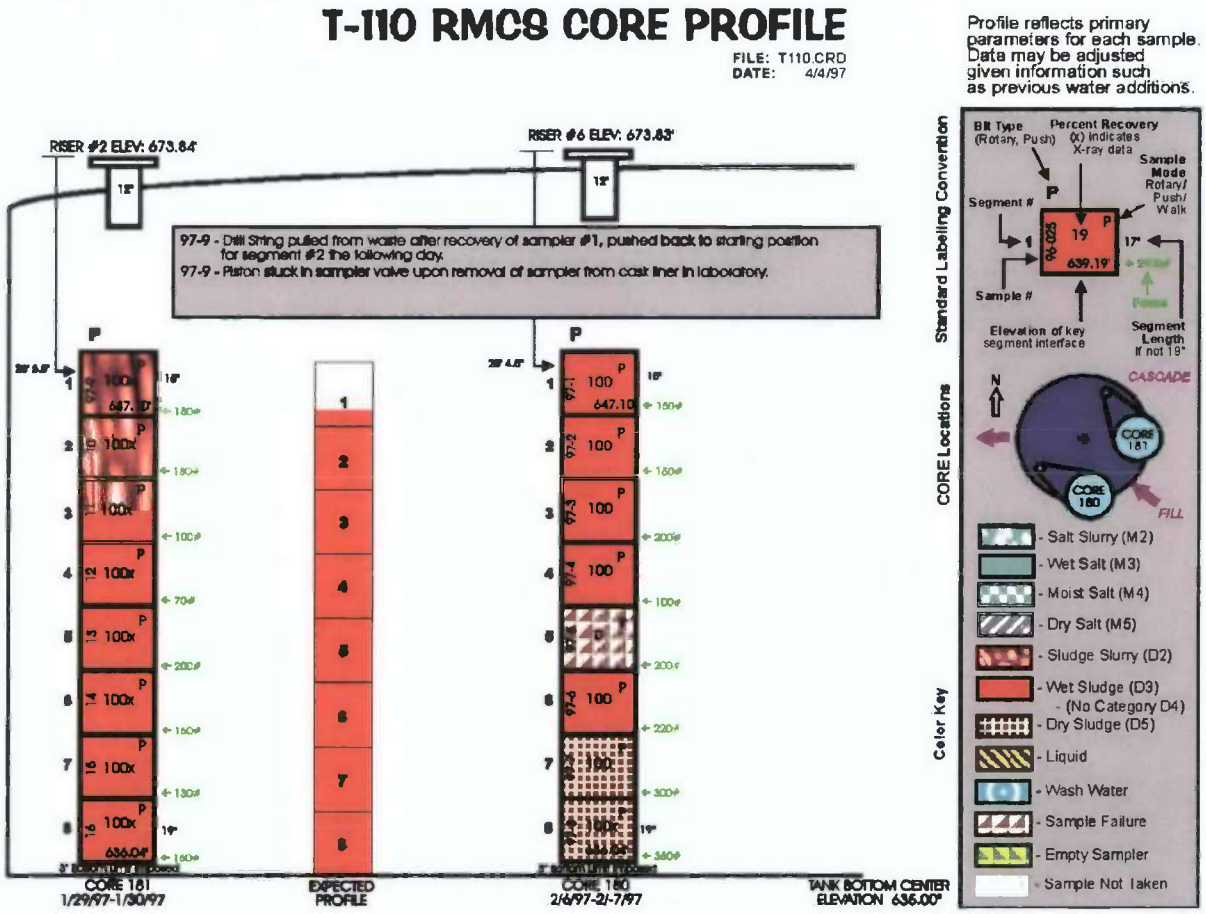
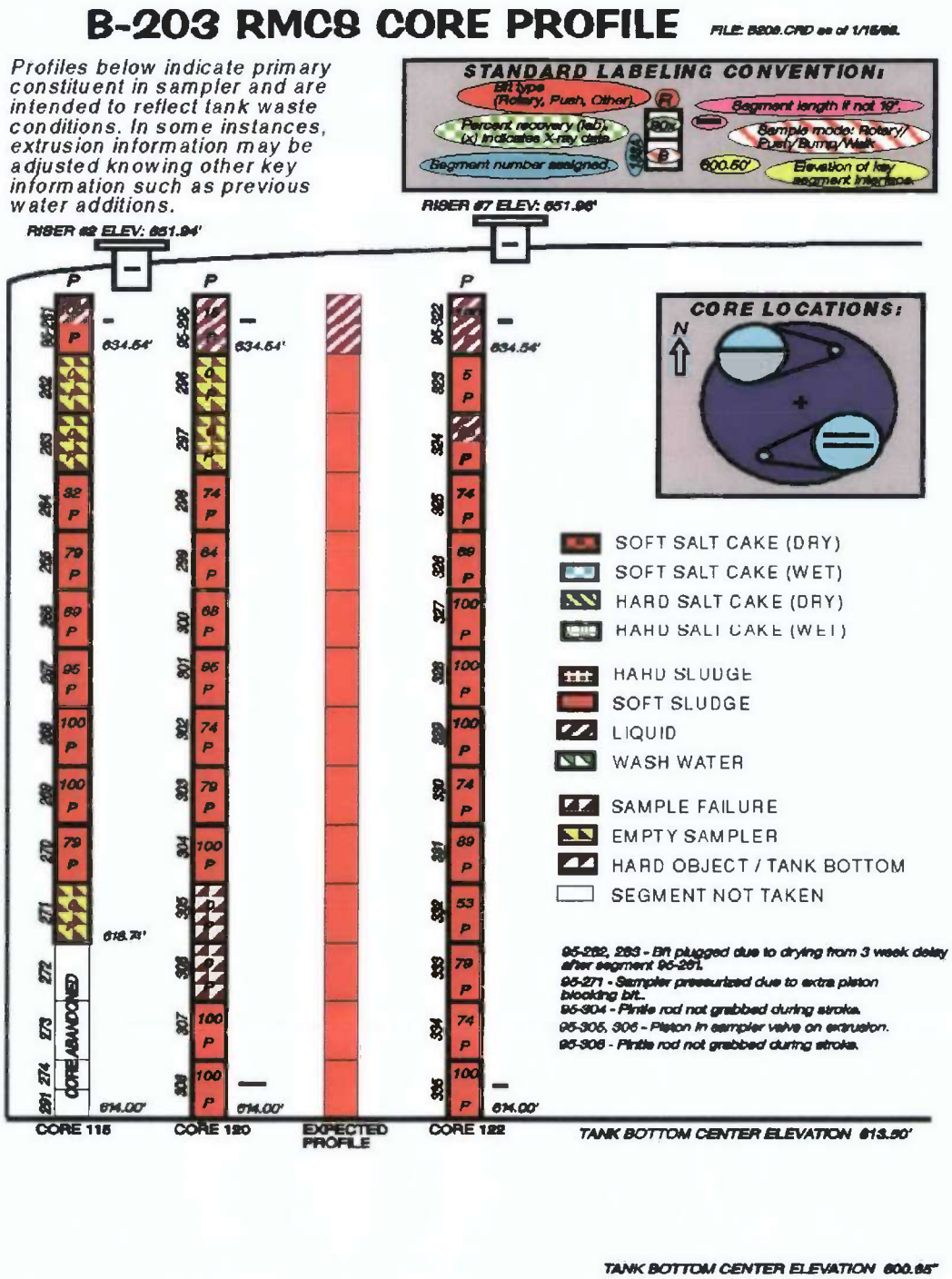


Figure A.1. Tank T-110 Core Profile



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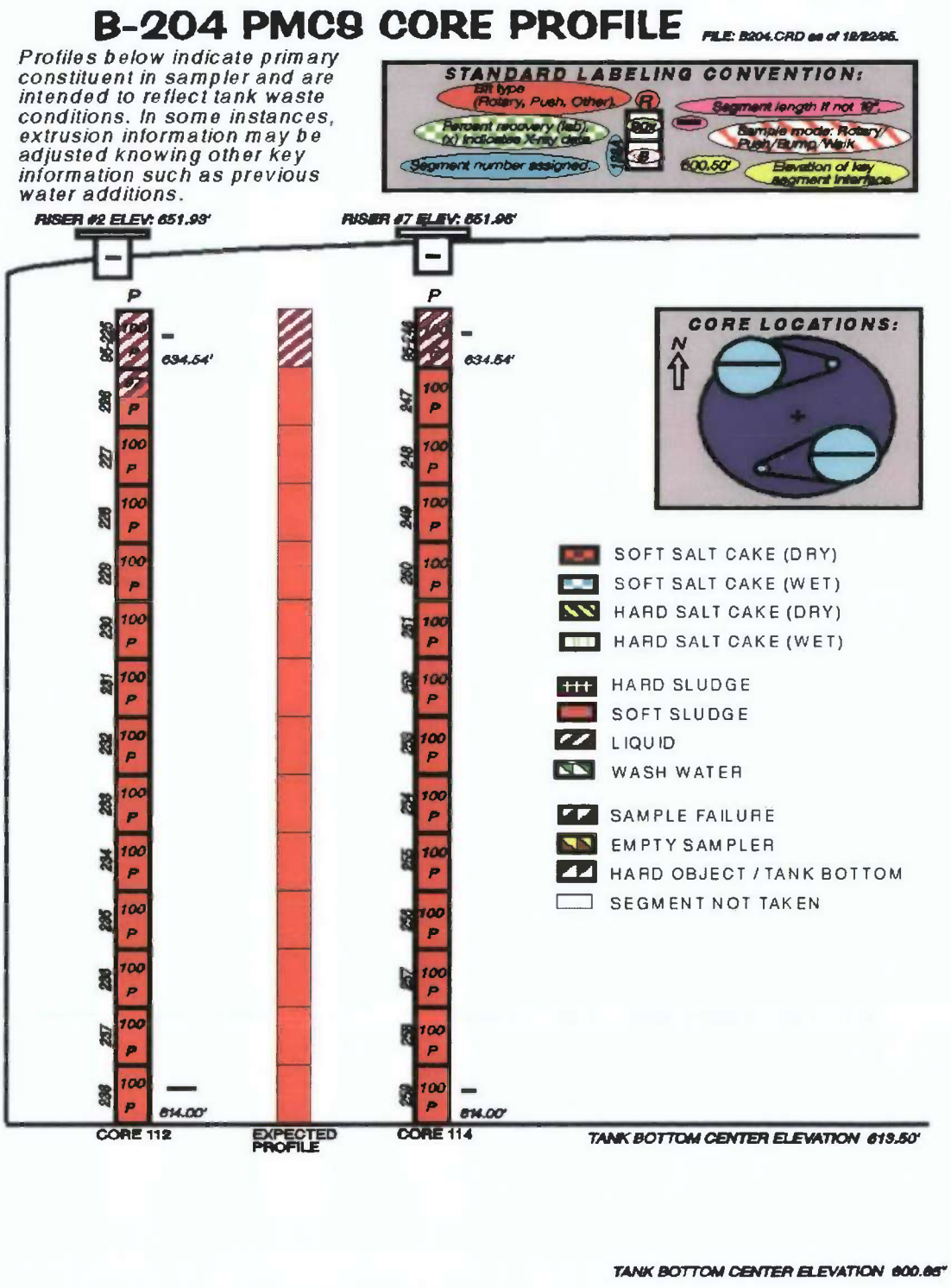

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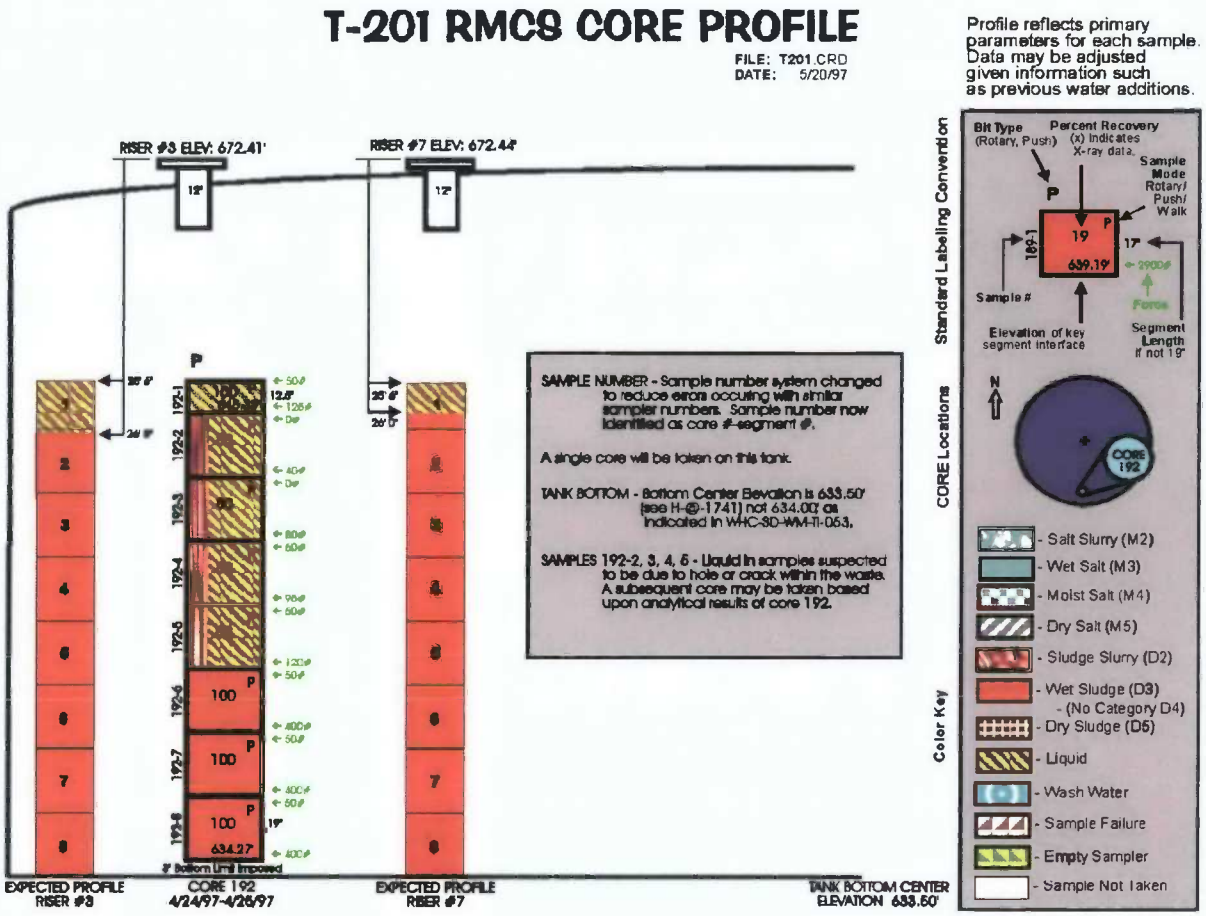

Figure A.4. Tank T-201 Core Profile

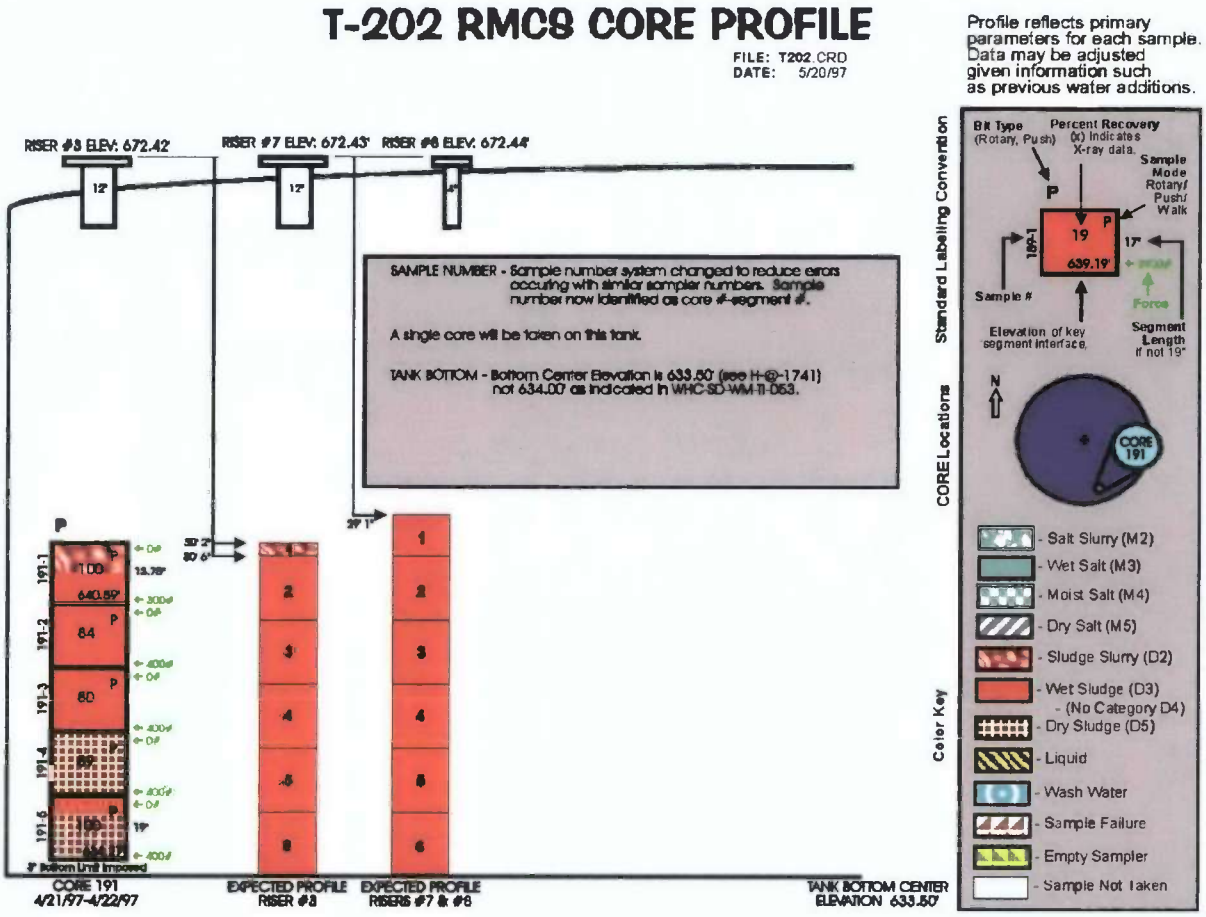

Figure A.5. Tank T-202 Core Profile

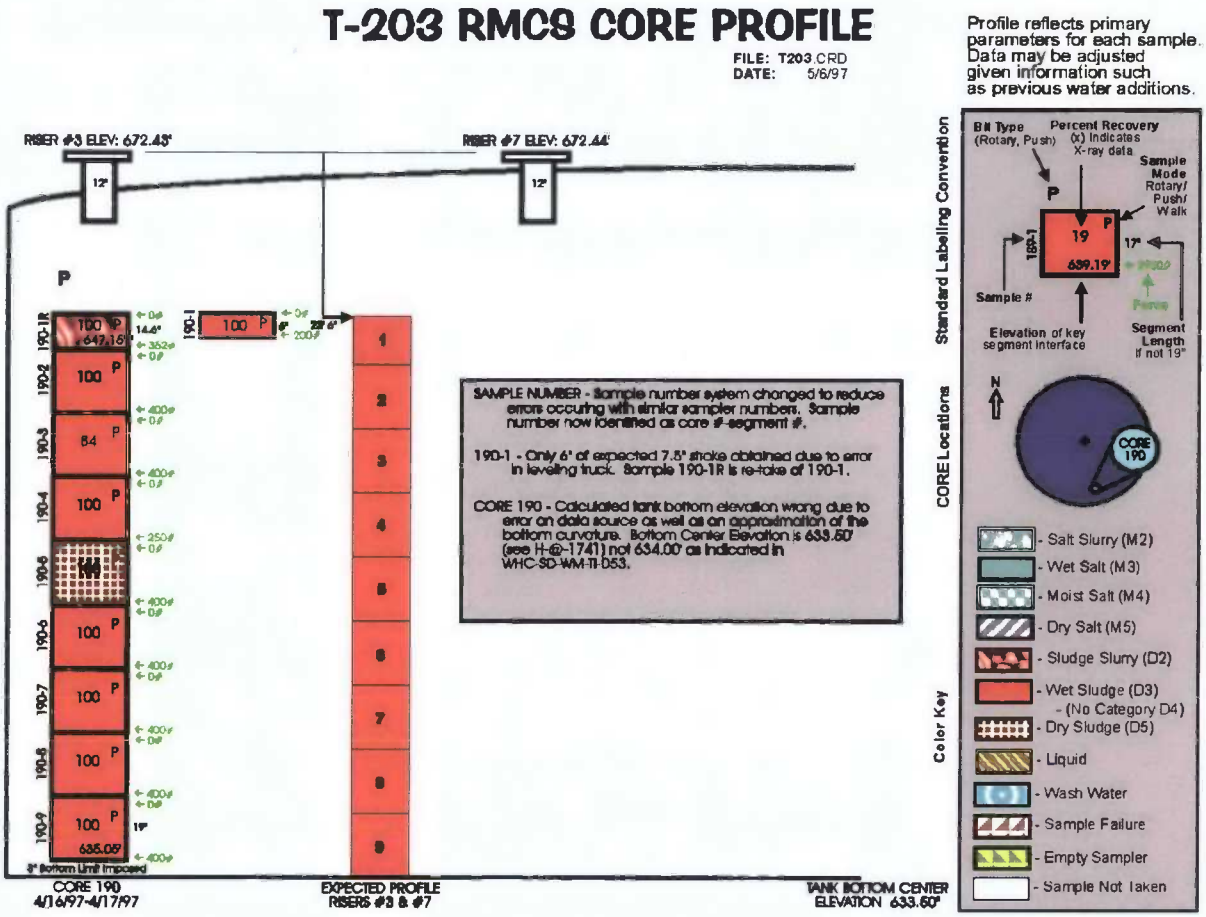

Figure A.6. Tank T-203 Core Profile

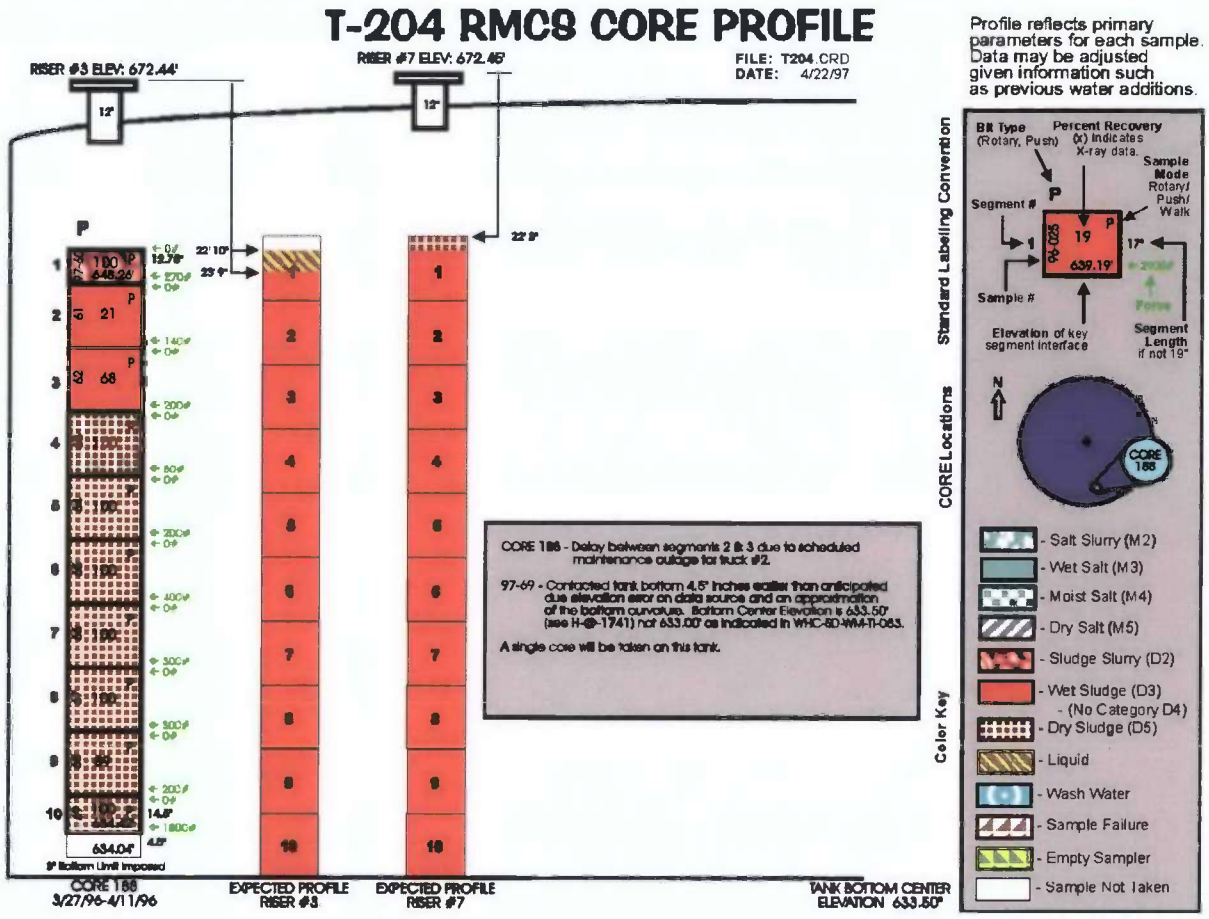

Figure A.7. Tank T-204 Core Profile

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July 16, 2009

CHPRC-0900932 R1

Mr. William J. Johnson, President
Washington River Protection Solutions, LLC
Post Office Box 850
Richland, Washington 99352

Dear Mr. Johnson:

RESPONSE TO REQUEST FOR EXCEPTION TO HNF-EP-0063, APPENDIX I, FOR NEWLY GENERATED TRANSURANIC WASTE FROM WASHINGTON RIVER PROTECTION SOLUTIONS

Reference: Letter, J. A. Nielsen, WRPS, to G. C. Triner, M&EC, "Regarding Waste Profile WRPS-270-0001, RH Mixed TRU Waste," WRPS-0900932, dated June 8, 2009.

In regard to the reference letter from Washington River Protection Solutions (WRPS), Waste Support Services, Technical Services, in consultation with the TRU Program, Nuclear Safety, Fire Protection and the U.S. Department of Energy-RL, has reviewed the request for the exception submitted for HNF-EP-0063, Revision 14, *Hanford Site Solid Waste Acceptance Criteria*, Appendix I. It has been determined that the acceptance of the C104 Heel Jet Pump in a Department of Transportation IP-1 container (40 foot Conex) under the exception to HNF-EP-0063 would not affect compliance with any applicable regulations or any of the externally approved requirements of the Solid Waste Operations Complex Treatment Storage and Disposal unit.

An exception has been granted for the C104 Heel Jet Pump assessed as TRU coming for temporary storage at the Central Waste Complex based upon the following conditions:

- WRPS is responsible for all costs associated with the temporary storage, treatment, disposition and associated transportation costs.
- Temporary storage will last no more than six months.

Should you have any questions or concerns, please contact D. J. Moak of my staff on 373-1174 or G. C. Triner on 373-3173.

Sincerely,

John G. Lehew III
President and
Chief Executive Officer

GCT/lgj

cc: M. S. Collins, RL
J. A. Nielsen, WRPS

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Basis for Designating Certain Hanford Single-Shell Tank Waste Resulting from the Bismuth-Phosphate Process as Transuranic Waste

Prepared for the U.S. Department of Energy
Assistant Secretary for Environmental Management

Office of River Protection
P.O. Box 450
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W. M. Hewitt, YAHSGS LLC

February 2004

Prepared for the U.S. Department of Energy
Assistant Secretary for Environmental Management

Office of River Protection
P.O. Box 450
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DOE/ORP-2004-01

Basis for Designating Certain Hanford Single-Shell Tank (SST) Wastes Resulting from the Bismuth-Phosphate Process (BPP) as Transuranic Waste (TRU)

January 2004

Prepared by YAHSGS LLC, Richland, WA
under Contract ORP-YAH001 to the
Project Assistance Corporation in Support of the
Department of Energy, Office of River Protection

Preface

Although historically the Department of Energy (DOE) has managed wastes within the Hanford tank farms as High-Level Wastes (HLW) as a matter of operations management policy, DOE has long maintained that, based on origin, process history, and radiological characteristics, the wastes in any specific tank may actually be HLW, Transuranic Waste (TRU), or Mixed Low-Level Waste (MLLW). DOE, therefore, has planned to appropriately designate wastes into one of those categories once the wastes are ready for retrieval for treatment and disposal.

Accordingly, the DOE Office of River Protection (ORP) identified 11 Single-Shell Tanks (SSTs) that contain wastes from the Bismuth-Phosphate Process (BPP). The BPP, the first production-scale Spent Nuclear Fuel (SNF) reprocessing process ever used, was deployed during the Manhattan Project (World War II) to separate plutonium from SNF. The BPP was only used at Hanford and was replaced 50 years ago by more efficient solvent extraction reprocessing processes, i.e., Reduction and Oxidation (REDOX) and Plutonium-Uranium Extraction (PUREX). An important feature of the BPP relative to waste designation is that it was a batch process, a feature that allows ORP to clearly distinguish where SNF existed (or did not exist) within the process. The BPP used chemical additions to selectively dissolve and precipitate plutonium compounds so that the plutonium could be separated from other SNF constituents by liquid/solids separations via centrifugation. Multiple water washes, each followed by centrifugation, ensured very high degrees of solids separation from process liquids, e.g., separation of plutonium precipitates from liquids produced directly in SNF reprocessing.

The BPP created HLW that will be treated in the Waste Treatment Plant currently under construction at Hanford and subsequently disposed of in the national repository. The BPP also produced waste streams that are not HLW by origin as those wastes were not produced during the reprocessing of SNF. The fact that the wastes are not HLW is confirmed by waste fission product concentrations that are orders of magnitude less than those the U.S. Nuclear Regulatory Commission requires to be disposed of in a geologic repository (10 CFR Part 61, Low-Level Radioactive Waste Disposal).

This document explains the BPP and identifies which BPP steps produced HLW and which did not on the basis of where SNF reprocessing actually took place within the series of BPP batch treatment steps. As a result, this document provides a technical and regulatory basis for DOE-ORP to determine that wastes from the BPP that are now contained in 11 Hanford SSTs (B-201, B-202, B-203, B-204, T-201, T-202, T-203, T-204, T-104, T-110, and T-111) are TRU due to waste origin and confirmed by radionuclide content. This document was developed in full consideration of extensive technical evaluations of historical BPP and tank farm source documents and records that were performed by the current Hanford tank farm contractor, CH2M HILL Hanford Group, Inc. (CH2M HILL). CH2M HILL's evaluations included historical records and process information produced by Hanford site contractors that operated the BPP over its 1945–1954 operating history. Information derived from those historical documents is consistent with the radioactive and chemical characteristics of the wastes in the 11 SSTs. Accordingly, this document is believed to provide a reasonable and sound basis to support a DOE-ORP determination that the wastes in the 11 SSTs identified above are TRU. Once those wastes are put into a suitable form for disposal, appropriately packaged, and characterized in a

manner that conforms to the Waste Isolation Pilot Plant (WIPP) waste acceptance criteria and permit requirements, those wastes should be suitable for disposal at WIPP.

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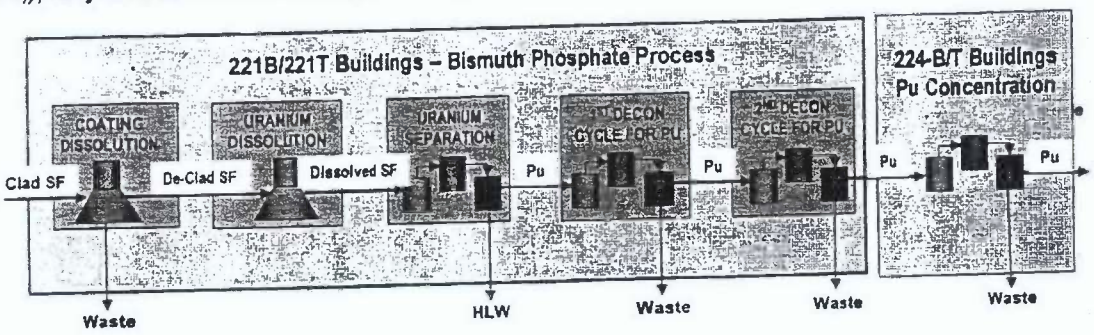
Executive Summary

The diverse nature of Hanford's tank waste generation operations over the past 60 years has led to large tank-to-tank differences in radioactive material concentrations. Understanding how and why these differences occurred is important to reaching sound waste management decisions. Of particular interest are wastes generated from the Bismuth Phosphate Process (BPP), the first ever Spent Nuclear Fuel (SNF) reprocessing and Plutonium (Pu) recovery process. That is, in part, because wastes generated by several BPP process steps are candidates for a Transuranic Waste (TRU) determination as illustrated and discussed below.

The BPP, unlike later Hanford solvent extraction-based reprocessing approaches (i.e., REDOX and PUREX), consisted of a series of individual batch processes which selectively dissolved and precipitated specific materials to recover Pu. It achieved thorough liquid/solids separation via centrifugation and multiple water rinses of the centrifuge solids cake, thereby removing liquids and soluble materials from the cake. Each batch process step resulted in an extensive and selective separation of the process wastes from the process product streams. As a result, out of the five distinct BPP process steps (coating dissolution, U dissolution, U separation, 1st decontamination cycle for Pu, 2nd decontamination cycle for Pu,), only two involved SNF reprocessing: U dissolution and U separations.



The coating removal process did not create High-Level Waste (HLW) because it only dissolved the aluminum coating leaving the SNF intact. That process did not dissolve SNF and its wastes were mildly contaminated.

HLW including all liquids produced directly in the reprocessing of SNF existed only within the U dissolution and U separation processes. Acids introduced during U dissolution dissolved the SNF, placing the Pu, the U, and all of the fission products in solution. The U separation processes then selectively precipitated the Pu, leaving the U and fission products in solution.

The liquid waste from U separations contained over 99.5% of the SNF constituent elements including >99.5% of the U, ~99% of the Cs-137, and ~90% of the Sr-90 (DuPont 1944). The liquid and solid wastes produced during U dissolution and U separation therefore fall squarely within the definition of HLW as set forth in the Nuclear Waste Policy Act of 1982 (NWPA). The extensive liquid/solids separations and multiple rinses conducted during U separations assured that any liquid wastes produced directly in reprocessing were discharged as liquid wastes and did not follow the Pu precipitate into the 1st or 2nd decontamination cycles or beyond.

The Pu precipitate, once triple rinsed, contained >99.5% of the Pu, <0.5% of the U, and ~10% of the fission products. At least half of the fission products were short-lived isotopes that decayed to deminimis levels within 1-2 years. Because the SNF constituent elements were separated during U separations, no SNF was present in the subsequent decontamination cycles. Accordingly, wastes from the 1st and 2nd decontamination cycles and Pu concentration process are not HLW based on the NWPA HLW definition.

The low fission product concentrations in those wastes is consistent with a non-HLW designation. Therefore, on the basis of origin and content, the wastes in the 11 SSTs that received the wastes from coating removal, the 1st and 2nd decontamination cycles, and Pu concentration (T-104, T-110, T-111, B-201 through B-204, T-201 through T-204) are not HLW.

Moreover, the wastes in those 11 SSTs meet the definition of transuranic waste set forth in the NWPA and the Waste Isolation Pilot Plant (WIPP) Land Withdrawal Act of 1996 and are, therefore, candidates for disposal at WIPP in New Mexico.

Basis for Designating Certain Hanford Tank Wastes as TRU

1.0 BACKGROUND – Hanford Wastes Vary Significantly Tank-to-Tank

Hanford's 149 SSTs, 28 Double-Shell Tanks (DSTs), and 60 Miscellaneous Underground Storage Tanks (MUSTs) collectively store ~54 million gallons of radioactive mixed defense wastes containing ~190 million curies of radioactivity. The wastes in those tanks have varying origins. For example, although extensive SNF reprocessing operations were conducted at Hanford, not all tank wastes originated during the reprocessing of SNF. Tank wastes were produced by a number of Hanford defense-related operations associated with removing cladding from SNF, purifying the Pu product, decontaminating equipment/facilities, and performing laboratory analyses. Rather than being the actual reprocessing of SNF, these operations occurred prior to, following, or incidental to SNF reprocessing. This diversity in Hanford's tank waste generation operations resulted in large tank-to-tank radioactive material concentration differences. Understanding these differences is important to sound waste management decisionmaking. The magnitude of the large tank-to-tank radionuclide concentration differences are graphically and numerically illustrated in Figures 1 and 2, respectively. For example, the five tanks[1] with the highest inventories of radioactive materials in Figure 1 collectively contain ~ 50 million curies whereas the 10 tanks[2] with the lowest radioactive material inventories collectively contain less than 5 thousand curies; this is a factor of 10,000 difference. Furthermore, specific radionuclide concentrations can vary by factors greater than 1 million from tank-to-tank as illustrated in Figure 2 for Cs-137 and Sr-90, the two most prominent radionuclides in the tanks.

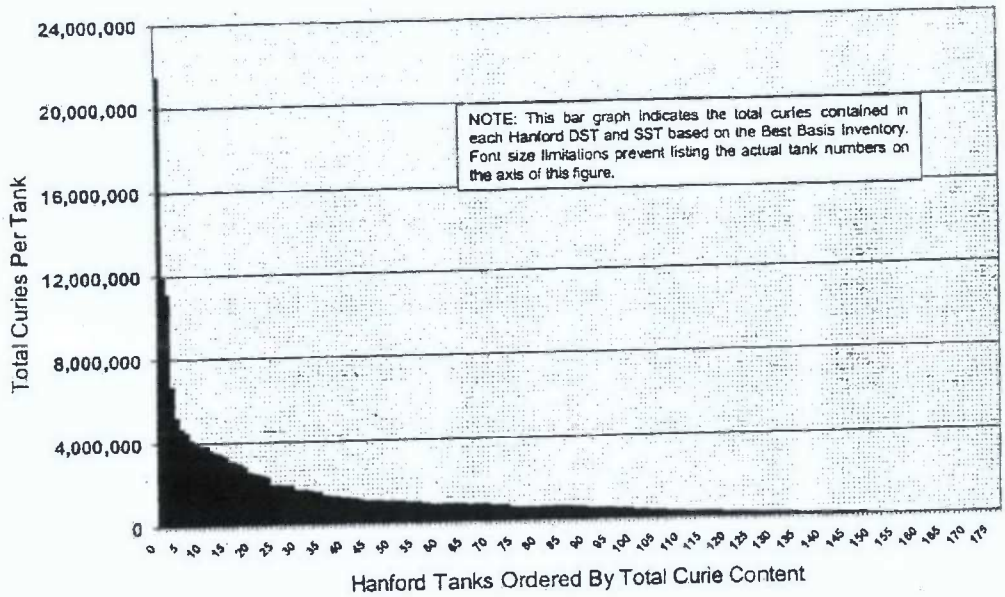


Figure 1. Radionuclide Inventories in the Hanford Tanks Span Over Four Orders of Magnitude

[1] In order of curie inventory, high to low, the tanks are AZ-101, AZ-102, AY-102, A-105, and AX-104.
[2] Not ordered by curie inventory, the tanks are B-201, -202, -203. -204; T-201, -202, -203, - 204; and U-203 and U-204.

Source: Best Basis Inventory in the TWINS Database

There are several reasons why there is such a wide range of fission product inventories in the Hanford tanks. First, while some tanks received highly radioactive wastes produced during the reprocessing of SNF, other tanks did not. Second, the BPP, the world's first production-level reprocessing process which was carried out at Hanford during the Manhattan Project starting in 1944, created large quantities of relatively low-curie waste compared to the waste produced by later, substantially more efficient processes such as REDOX and PUREX. Third, a 1960s/1970s Hanford tank waste campaign extracted large amounts of Cs-137 from liquids in most Hanford tanks and Sr-90 from wastes in the A and AX Farm tanks. That campaign reduced the collective Hanford tank farms fission product content by approximately 40%[3]. Fourth, tank capacities vary from 55,000 gallons to over 1 million gallons and tanks are filled to varying degrees.

	Cs-137		Sr-90	
	(Ci/liter)	Tank	(Ci/liter)	Tank
Highest Concentration	~1.9	AX-104	~79	AX-104
Lowest Concentration	0.00001	T-204	<0.000003	T-202
Ratio (High/Low)	200,000		30,000,000	

Figure 2. Highest and Lowest Cs-137 and Sr-90 Concentrations in Hanford Tanks
Source: Best Basis Inventory in Hanford TWINS Database

This variability in waste sources and concentrations has led DOE to consider the origin and the characteristics of wastes in each tank in planning its treatment and disposal strategies. Some examples of wastes discharged to tanks that did not originate directly during the reprocessing of SNF include:

- Decladding wastes resulting from dissolving the metallic cladding (coating) from the SNF in order to expose the actual fuel to reprocessing acids.
- Wastes from processes used to clean and/or concentrate recovered Pu product materials in order to achieve requisite Pu purity levels for weapons use.
- Laboratory wastes resulting from the sampling and analysis of various process and waste streams resulting from Hanford operations.
- Wastes from the cleanup of contaminated facilities and/or equipment.

Regardless of the characteristics or origin of the waste in any given tank, as a matter of policy, DOE manages the Hanford tank farm wastes as HLW while those wastes are stored in the tanks. This does not mean that DOE classified the wastes as HLW but rather, that DOE employs an appropriately conservative management practice to ensure that the highest levels of safety and best management practices are in place during the storage, retrieval, and handling of the Hanford tank farm wastes.

In the sections that follow, the BPP is described with a focus on determining (a) when SNF was present such that the "reprocessing of SNF" actually occurred in a process, (b) which BPP

[3] The cesium and strontium were converted to cesium chloride and strontium fluoride and encapsulated. The campaign was undertaken to reduce the decay heat load on the tanks, however, beneficial uses for the capsules were sought and many capsules were deployed on commercial and government initiatives.

processes created "liquid waste produced directly in reprocessing [of SNF]", and (c) which BPP processes appear to have resulted in solid materials with "fission products in sufficient concentrations" to warrant permanent isolation. The BPP is compared and contrasted as appropriate with the PUREX process for the simple reason that most people think of the PUREX process when they think of reprocessing. PUREX was used across the DOE weapons complex for Pu and Uranium (U) recovery. It was used in the United States on a limited basis for commercial reprocessing. Finally, PUREX is used internationally for commercial and defense reprocessing purposes (PNNL 1998). Conversely, the BPP was an earlier process used only at Hanford in the U.S. Government's first production-level campaigns to recover Pu for defense purposes. It processed less than 8% of the SNF reprocessed at Hanford.

2.0 BISMUTH PHOSPHATE PROCESS

As illustrated in Figure 3, the BPP[4] was carried out in 221-T plant from 1944 to 1956 and in 221-B plant from 1945 to 1952. As the first reprocessing process ever used at production levels to separate Pu from SNF, it was conceived with an emphasis on time and purpose rather than efficiency. The BPP was a batch process. It deployed a complex chemistry that selectively dissolved and precipitated targeted chemical compounds such that simple liquid/solids separations equipment (centrifuges) could isolate Pu from the other materials in the SNF as well as materials introduced in the BPP. To place the process in perspective, the Government's objective was to separate the one part Pu produced in the fission process from the roughly 10,000 parts of U and fission products that it was dispersed amongst in the SNF.

The BPP was quite different from successor reprocessing processes. For example, its sole purpose was to recover Pu. Uranium was discharged as a waste. Conversely, REDOX and PUREX recovered Pu and U, each as a separate product. Also, REDOX and PUREX were continuous solvent extraction processes which used a small fraction of the chemical additives that the BPP required for separations. As a result, the BPP created over 200 times more waste than PUREX per ton of U fuel processed. The BPP U Separations process created approximately ~3800 gallons of HLW per ton of U (GE 1951) while PUREX created ~40 gallons per ton (ARHCO 1968). This resulted in Hanford's PUREX wastes having substantially higher fission product concentrations than BPP wastes. For example, wastes discharged from the BPP U Separations process, the BPP waste stream with the highest fission product concentrations, were reported to have Cs-137 concentrations of approximately 60 Ci/m^3 (GE 1955), < 0.5% of the 13,000 Ci/m^3 Cs-137 concentrations in PUREX 1st

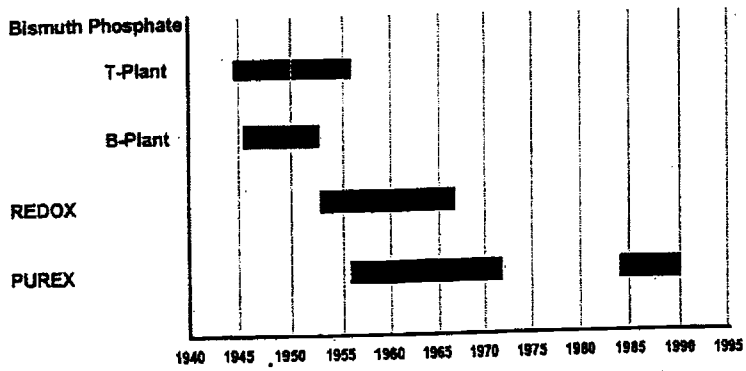


Figure 3. Operating Time Frames for Spent Nuclear Fuel Reprocessing Processes at Hanford

[4] The BPP flowsheets are provided in Attachment A and comparisons between the BPP and the PUREX process wastes are provided in Attachment B.

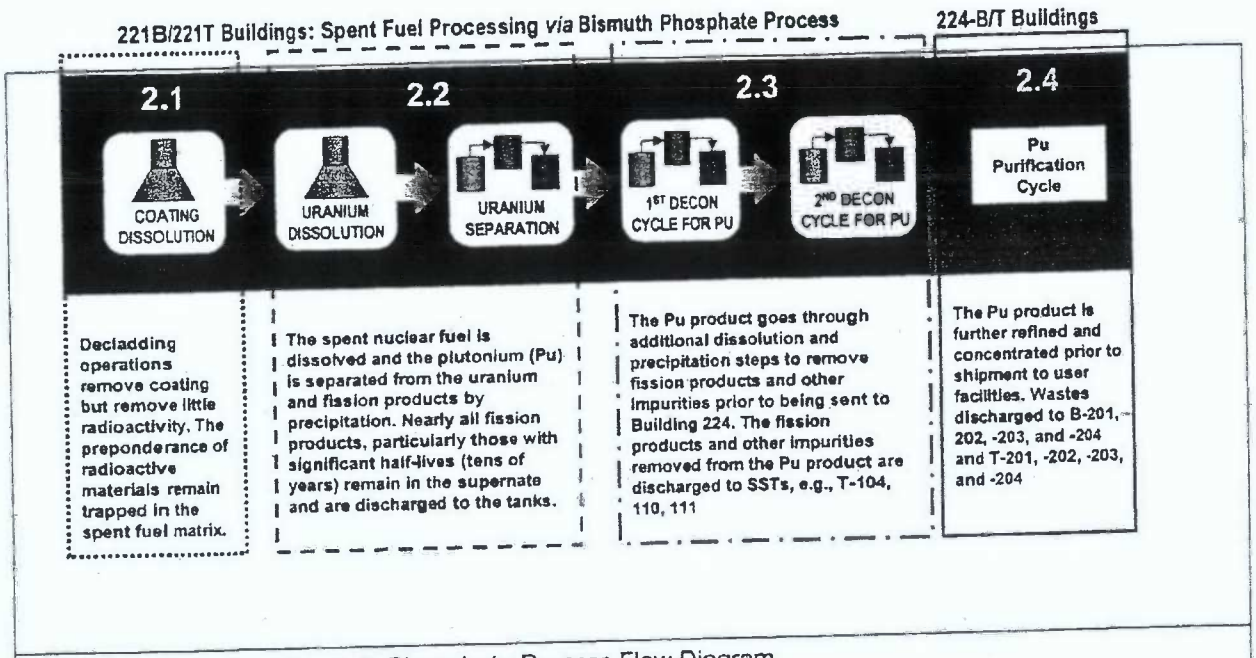


Figure 4. Generalized Bismuth Phosphate Process Flow Diagram
Source: Johnson 2003.

cycle raffinate wastes after neutralization (ARHCO 1968).

Figure 4 depicts the major BPP steps. The discussion that follows traces the SNF, the Pu product, and the process wastes through the BPP [Note that the numbering of the subsections that follow correspond to the numbers within each outlined block in Figure 4]. The following discussions include general information regarding the chemical processes used. More detail regarding the BPP chemistry and mass flow information can be found in Attachment A.

2.1 Coating Dissolution (Decladding – Figure 4, Block 2.1)

Prior to the actual reprocessing of SNF, the aluminum cladding (or coating) had to be removed to expose the U to the acids that would be used to dissolve it. A boiling sodium nitrate/sodium hydroxide solution was used to dissolve cladding. While virtually all of the radioactive fission products remained within the intact spent fuel matrix, small amounts of radioactive materials at the surface of the fuel slugs entered decladding solutions. Decladding operations are considered a "head end" process and not part of SNF reprocessing since the SNF remained intact throughout the decladding process. The decladding wastes were subsequently combined with 1st cycle Pu decontamination waste (discussed in Section 2.3) to use the excess sodium hydroxide in the decladding wastes to neutralize acids in the 1st cycle decontamination wastes.

2.2 Uranium Dissolution and Uranium Separation (Figure 4, Block 2.2)

Following decladding, the U fuel slugs were dissolved in nitric acid. Once dissolved, water and sulfuric acid were added to convert the uranyl nitrate to uranyl sulfate. Next, bismuth nitrate and phosphoric acid were then added and a bismuth phosphate carrier was formed that extracted Pu from solution as a precipitate. The uranyl sulfate remained in solution along with nearly all of the cesium and approximately 90% of the strontium (CH2M HILL 2002). The bismuth phosphate carrier and Pu were then precipitated as a filter cake via centrifuging, the filter cake

was rinsed with water and re-centrifuged three times to remove any waste liquids and soluble fission products that may have been initially entrained in the filter cake, and then the Pu cake was transferred to the first Pu decontamination cycle (GE 1951).

Approximately 10% of the fission products that were dissolved with the U stayed with the Pu cake when it moved from U separations to the first Pu decontamination cycle. In addition to strontium, substantial quantities of short-lived[5] fission products, such as zirconium-95 (Zr-95) and niobium-95 (Nb-95), were co-precipitated.

2.3 Plutonium Decontamination (Figure 4, Block 2.3, 1st and 2nd Decon Cycles)

In the first Pu decontamination cycle, the Pu was oxidized to the +6 valence state via the addition of sodium bismuthate and sodium dichromate. Sodium bismuthate, phosphoric acid, zirconium nitrate, and cerium nitrate were added to precipitate bismuth phosphate and fission products (primarily strontium, cerium, and zirconium). The bismuth phosphate and fission product precipitate were centrifuged to separate them from the Pu which remained in the liquid phase. Following separation, the Pu in the liquid phase was reacted with bismuth subnitrate and phosphoric acid to produce a bismuth phosphate carrier and co-precipitate plutonium phosphate. The bismuth phosphate carrier and plutonium phosphate solids were separated from the liquids by centrifugation. The plutonium phosphate solids were water washed and centrifuged three times. The bismuth phosphate and plutonium phosphate solids were then dissolved in nitric acid, forming plutonium nitrate and bismuth nitrate in solution. This solution was then transferred to the second decontamination cycle where the first decontamination process steps (except for zirconium nitrate and cerium nitrate addition) were repeated to further purify the Pu product.

2.4 Plutonium Concentration Building (224-B/T) Wastes (Figure 4, Block 2.4)

The Pu from 221-B/T plants was transferred to the 224-B/T Pu Concentration Building to remove the bismuth phosphate and residual fission products which were essentially all short half-life contaminants. The Pu solution was received at 224-B/T in a +4 valence state. It was first oxidized with sodium bismuthate to a +6 valence state. Phosphoric acid was added to precipitate bismuth phosphate along with residual Zr-95 and Nb-95 fission products, which were then removed by centrifugation leaving the Pu in solution. Hydrogen fluoride and lanthanum fluoride were added to precipitate remaining fission products leaving the Pu in solution. Hydrogen fluoride and lanthanum salts were then added to create lanthanum fluoride and plutonium fluoride solids which were separated by centrifugation. The lanthanum fluoride and plutonium fluoride solids were reacted with potassium hydroxide to produce lanthanum hydroxide and plutonium hydroxide. The lanthanum hydroxide and plutonium hydroxide solids were reacted with nitric acid to produce the high purity Pu nitrate/lanthanum nitrate product.

Targeted radionuclides for removal were primarily short-lived fission product and daughter isotopes of zirconium, cerium, lanthanum, ruthenium, praseodymium, and yttrium (DuPont 1945), many of which were difficult to physically separate from the Pu via precipitation

[5] Zr-95 has a 64-day half-life and Nb-95 a 35-day half-life. In addition to the Zr-95, other phosphate insoluble short-lived fission products such as Ce-144 (~284 days) were removed to achieve the desired plutonium purity and handling characteristics. The fission products of concern relative to long-term waste management and disposal are Cs-137 (~30 years) and Sr-90 (~29 years) which together with their daughters, Ba-137m and Y-90, account for ~99% of the curies in the Hanford tanks at the present time.

processes. Thus, multiple precipitation steps were used in the first and second Pu decontamination cycles and the Pu Concentration Building to separate these short-lived fission products from the Pu product.

3.0 CLASSIFICATION OF TANK WASTES FROM THE BISMUTH PHOSPHATE PROCESS

Although the BPP is referred to using the generic term 'reprocessing', the BPP actually consisted of batch chemical process operations. Unlike the later solvent extraction processes (REDOX, PUREX) which were continuous flow and continuously connected, each operation within the BPP took place on a batch basis. Figure 5 illustrates a typical BPP process step. Feed material enters a process tank. The feed could consist of a re-dissolved solids (such as SNF or a Pu cake) from a centrifuge or it could be the liquid phase from a centrifuge as illustrated in Figure 5. In either case, chemical additives (such as those listed in Section 2) are used to selectively keep certain chemical species in solution and to precipitate other species. The mixture is then transferred to a centrifuge where the solids are separated from the liquids by centrifugal force. The liquids are discharged from the centrifuge as it spins and the solids are retained. The tank where the feed and additives were mixed is then rinsed with water to ensure all precipitates are removed. Clean rinse water is sprayed onto the solids in the centrifuge (~3 parts water to 1 part solids) while it operates to replace any process liquids that may have been entrained in the solid cake. The centrifuge is operated two cycles to de-water the cake. Water is again sprayed onto the solids in the centrifuge in a second cake rinse (~3 parts water to 1 part solids) while it operates to wash trace quantities of dilute process liquids from the solid cake. The centrifuge is operated two cycles to de-water the cake. All liquids including rinses pass on to the next process step or are discharged as a waste based on the specific process operation. The solids are dissolved and then transferred to the next BPP process or discharged as a waste, again based on the specific BPP process operation.

In the manner discussed above, each BPP batch process achieved a highly effective liquid/solids separation without cross contamination between batch operations.

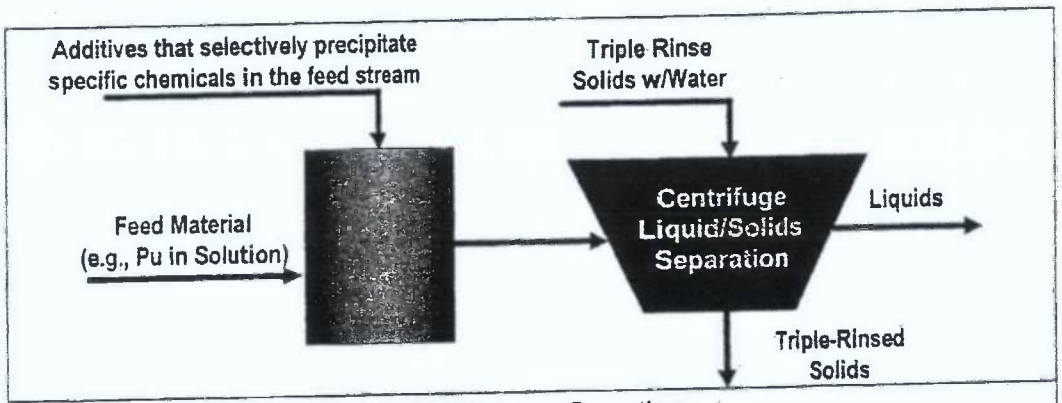


Figure 5 – Typical Bismuth Phosphate Process Operation

The clean separation liquid/solid separations and distinct break between BPP operations provides an ability to clearly demark where reprocessing of SNF did and did not occur, where "liquid waste produced directly in reprocessing" was present and where it was not, and consequently,

which BPP process operations created HLW and which did not. The process logic is described below.

3.1 Where Did SNF Reprocessing Occur?

SNF reprocessing could only occur during BPP process steps where the SNF constituent elements existed in solution. That is because the NWPA defines SNF as "fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated by reprocessing."

Based on that definition, the reprocessing of SNF in the BPP occurs during the U dissolution and U separation processes[6] as illustrated in Figure 6. The U dissolution and U separation processes are the only points along the BPP flowsheet where all of the constituent elements of the SNF existed in one place. The U dissolution process places the SNF constituent elements (U, Pu, fission products) into solution. All of the constituent elements of SNF exist at that point. The U separations process then selectively precipitates the Pu. All of the SNF constituent elements are present in the mixing tank and in the centrifuge.

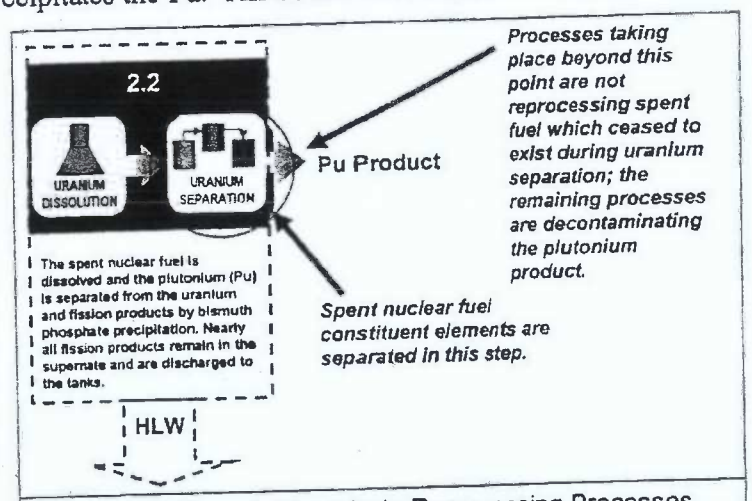


Figure 6. Bismuth Phosphate Reprocessing Processes

Once liquid/solids separations occur in the U separations centrifuge, the SNF constituent elements are separated into waste and Pu product cake. At the completion of the Pu product cake water rinses in the centrifuge, the constituent elements of the SNF have been fully separated and reprocessing is complete. The resultant waste and product streams are as follows:

- Uranium Separations Liquid Waste Stream – This waste stream includes ~99.5% (by mass) of all materials present in the SNF prior to dissolution including ~99.5% of the U, ~90% of all fission products including ~99% of the Cs-137 and ~90% of the Sr-90, a small fraction of the Pu, and chemicals/acids used to keep those materials in the liquid phase (CH2M HILL 2002, Johnson 2003), and
- Plutonium Product Cake – The Pu product cake includes the precipitated Pu, ~0.5% of the U, and ~10% of the fission products, at least half of which are short-lived fission products and daughters (Johnson 2003).

[6] Before uranium dissolution, reprocessing cannot occur since the SF constituent elements could not be separated by reprocessing while still in solid form.

3.2 Which Liquid Wastes Were Produced Directly In Reprocessing?

As described above 'liquid waste produced directly in reprocessing' could only have been created during U dissolution and U separations as those two BPP process steps were the only steps where reprocessing took place. The liquid wastes produced directly in reprocessing were separated from the Pu product by centrifugal action.

The Pu product stream was thoroughly rinsed and centrifuged multiple times to remove all traces of the liquids produced directly in reprocessing (and the undesirable contaminants contained in such liquids) from the Pu cake. By the time the cake was transferred to the first Pu decontamination cycle, any residual liquids produced directly in reprocessing that remained in the cake would have been diluted by ~1000:1 and would have represented <0.1% of the volume of liquid created during U dissolution and U separations[7], a negligible volume and concentration. This, the, leads one to the conclusion that the only 'liquid waste produced directly in reprocessing' from the BPP is the liquid waste stream discharged from the U separations process to the SSTs.

3.3 Which BPP Wastes Are HLW?

For the BPP, it is evident from the preceding discussions that the liquid waste stream discharged from the U separations process contained "highly radioactive material resulting from the reprocessing of spent nuclear fuel". Those wastes therefore meet the definition of HLW set forth in the NWPA[8]:

> "High-level radioactive waste means:
>
> (A) the highly radioactive material resulting from the reprocessing of SNF, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; and
>
> (B) other highly radioactive material that the NRC, consistent with existing laws, determines by rule requires permanent isolation."

The U separations liquid waste stream is therefore identified in Figure 6 as HLW. That waste stream contained approximately 95% of the fission products important to DOE in determining the waste disposal pathway, i.e., ~99% of the Cs-137 and ~90% of the Sr-90, the two fission products that, with their secular equilibrium daughters, account for 99% of the radioactivity in the Hanford tanks[9].

---

[7] Cake volume approximately 10 gallons, moisture content ~30%. Waste from U separations approximately 2400 gallons (GE 1951). On that basis, (10)(0.3)/2400 = 0.1% of liquids produced directly in reprocessing should remain in the cake after first liquid/solid separation. Each rinse used 30 gallons of water (GE 1951). Assuming 3 gallons of liquid in the cake (30%) and three separate 30 gallon rinses (including tank rinse), each rinse should reduce the concentration by a factor of 10. Moreover, any such liquid would be highly diluted (by a factor of 1000 due to the three rinses) before the cake was dissolved and transferred.

[8] This same definition is incorporated by reference into the Atomic Energy Act of 1954 (AEA), as amended, and the Waste Isolation Pilot Plant Land Withdrawal Act.

[9] Ba-137m and Y-90 are daughters of Cs-137 and Sr-90, respectively, that are in secular equilibrium, i.e., the half-life of the parent radioisotopes (Cs-137 and Sr-90) is so much longer than that of the daughters that the radioactivity of the daughters is essentially equal to that of the parent.

The liquid wastes produced directly in reprocessing are part of that waste stream and were not present in the BPP Pu-related processes that followed U separations.

Accordingly, wastes from the BPP 1st and 2nd decontamination cycles are not HLW. Similarly, wastes from Pu concentration activities that further processed the product stream from the BPP in 224-B/T buildings were also not HLW.

4.0 TRU DETERMINATION – Candidate Wastes for Classification as Contact-Handled TRU

The WIPP Land Withdrawal Act defines TRU as:

> "waste containing more than 100 nanocuries of alpha-emitting transuranic isotopes per gram of waste, with half-lives greater than 20 years, except for (A) HLW; (B) waste that the Secretary has determined, with the concurrence of the Administrator, does not need the degree of isolation required by the disposal regulations; or (C) waste that the NRC has approved for disposal on a case-by-case basis in accordance with Part 61 of title 10, Code of Federal Regulations".

The waste streams from the BPP first and second decontamination cycles and the Pu Concentration Cycle that were carried out in the 224-B/T buildings are currently contained in 11 SSTs along with the decladding waste. Based upon the discussions in Section 3, none of those tanks contain HLW as defined in the NWPA.

Fission product concentrations[10] for the wastes in the 11 SSTs are illustrated in Figure 7. The

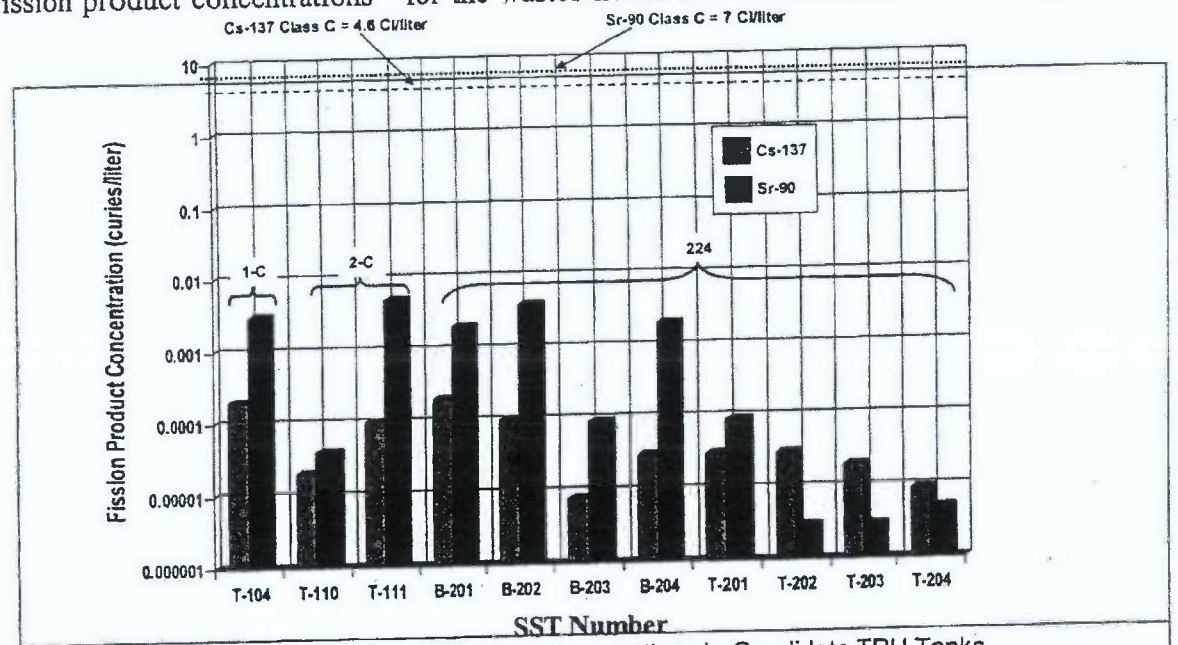


Figure 7. Cs-137 and Sr-90 Concentrations in Candidate TRU Tanks

[10] At the present time, Cs-137 and Sr-90 together with their daughters in secular equilibrium (Ba-137m and Y-90) represent ~99% of the fission product activity in the Hanford tanks (Best Basis Inventory in the Hanford TWINS database).

two dotted/dashed lines near the top of Figure 7 indicate the Class C concentration limits for Cs-137 (4.4 curies per liter) and Sr-90 (7 curies per liter)[11].

All 11 SSTs would be Class A or Class B solely on the basis of the 10 CFR § 61.55 concentrations related to fission products[12]. Based on the fission product content, DOE estimates that all 11 tanks will result in contact-handled[13] TRU once dewatered and packaged. The transuranic material content for each SST is indicated in Figure 8.

The tanks are grouped in Figure 8 according to the primary origin of their contained wastes from within or resulting from the BPP. The first eight tanks are all 200-series, 55,000 gallon, SSTs that contain 224-B/T Pu Concentration Building wastes.

Figure 8. Candidate Contact-Handled Single-Shell Tanks TRU Waste Designation

Tank	Waste Volume (kgal)	Waste Types (See Key Below)	TRU (nanocuries/gm)	Cs-137 (curie/liter)	Sr-90 (curies/liter)
Group I – Single-Shell Tanks Containing 224 Building Waste					
B-201	30	224	824	0.0002	0.002
B-202	29	224	214	0.0001	0.004
B-203	51	224	297	0.000008	0.00009
B-204	50	224	263	0.00003	0.0017
T-201	29	224	754	0.00004	0.0001
T-202	21	224	221	0.00003	0.000003
T-203	37	224	295	0.00002	0.000003
T-204	37	224	243	0.000009	0.000005
Group II – Single-Shell Tanks Containing 224 Building Waste and 2nd Decontamination Cycle Waste					
T-110	370	224/2C	67 (170 after drying)	0.00002	0.00004
T-111	447	224/2C/DW	182	0.0001	0.005
Group III – Single-Shell Tanks Containing 1st Decontamination Cycle Waste					
T-104	317	1C/CW	158	0.0002	0.003

KEY TO WASTE TYPE DESIGNATION

Waste Type	Description
1C	First Pu Decontamination Cycle Waste from Bismuth Phosphate Plant
2C	Second Pu Decontamination Cycle Waste from Bismuth Phosphate Plant
224	224-B/T Plutonium Concentration Building Waste
CW	Coating Removal Waste from Dissolution of the Coating on Spent Nuclear Fuel
DW	Equipment decontamination waste from 221-T Plant

[11] 10 CFR 61.55, Table 2. That regulation indicates the concentrations in curies per cubic meter. The Class C concentrations for Cs-137 and Sr-90 are 4400 curies per cubic meter and 7000 curies per cubic meter, respectively.
[12] The wastes exceed the Table 1 limits in §61.55 for alpha-emitting radionuclides, however, for defense wastes containing alpha-emitting radionuclides, the TRU definition in the WIPP Land Withdrawal Act are governing.
[13] Contact dose at the package surface will be less than 200 mR/hour.

The second group of tanks contain Pu Concentration Building wastes along with wastes from the BPP second decontamination cycle. T-111 also contains decontamination wastes.

The last group has one tank, T-104. It received BPP wastes from coating dissolution and the first decontamination cycle.

DOE has used historical information, sampling, and analysis to determine that the 11 SSTs identified in Figures 7 and 8 are valid candidates to receive a contact-handled TRU designation. That designation will be achieved through a ROD pursuant to the National Environmental Policy Act of 1969.

5.0 REFERENCES

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10 CFR Part 61, Licensing Requirements For Land Disposal Of Radioactive Waste, §61.55, Waste Classification.

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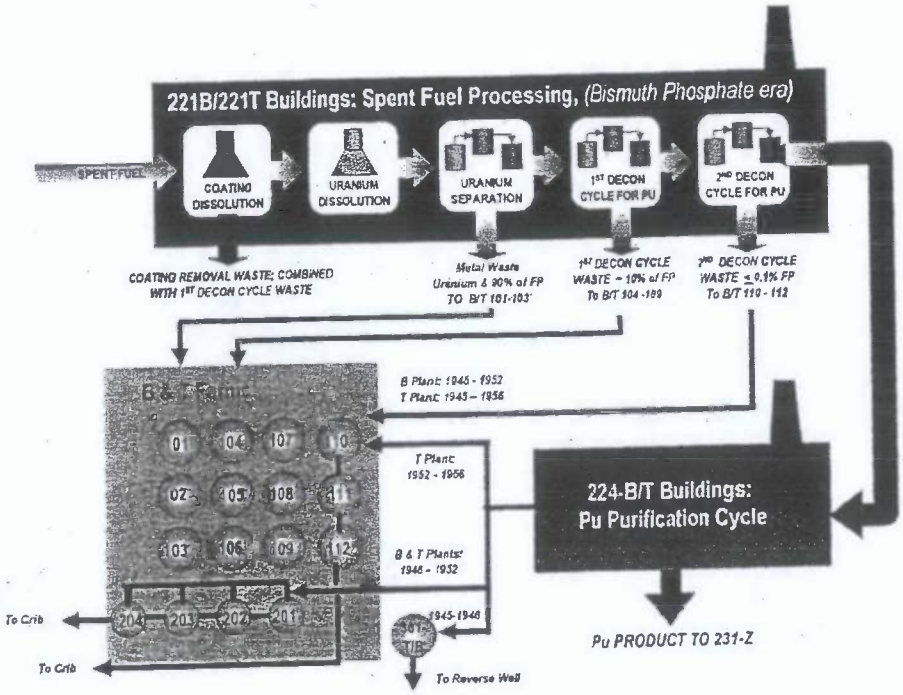
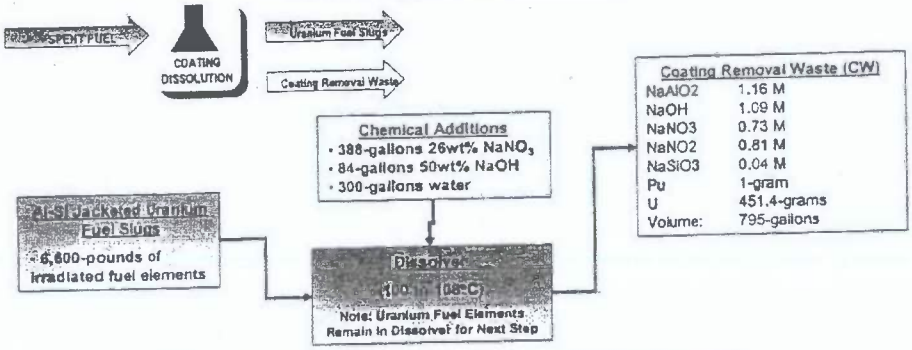
Best Basis Inventory (BBI), Tank Waste Information Network System (TWINS), http://twins.pnl.gov/twins.htm

APPENDIX A – Chemical Reactions for the Bismuth Phosphate Flow Sheet



Coating Dissolution

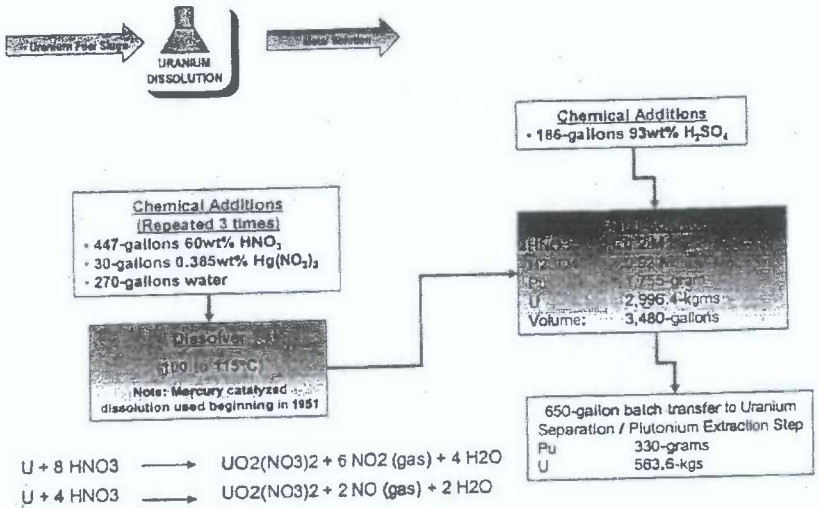


$$2\,Al + 2\,NaOH + 3\,NaNO_3 \longrightarrow 2\,NaAlO_2 + 3\,NaNO_2 + H_2O$$

$$8\,Al + 5\,NaOH + 3\,NaNO_3 + 2\,H_2O \longrightarrow 8\,NaAlO_2 + 3\,NH_3$$

NOTE: The BPP flowsheets in Appendix A were developed by Michael Johnson of CH2MHILL Hanford Group, Inc in December 2003 based upon his review of historical Hanford documents and records as identified at the end of this appendix.

Uranium Dissolution



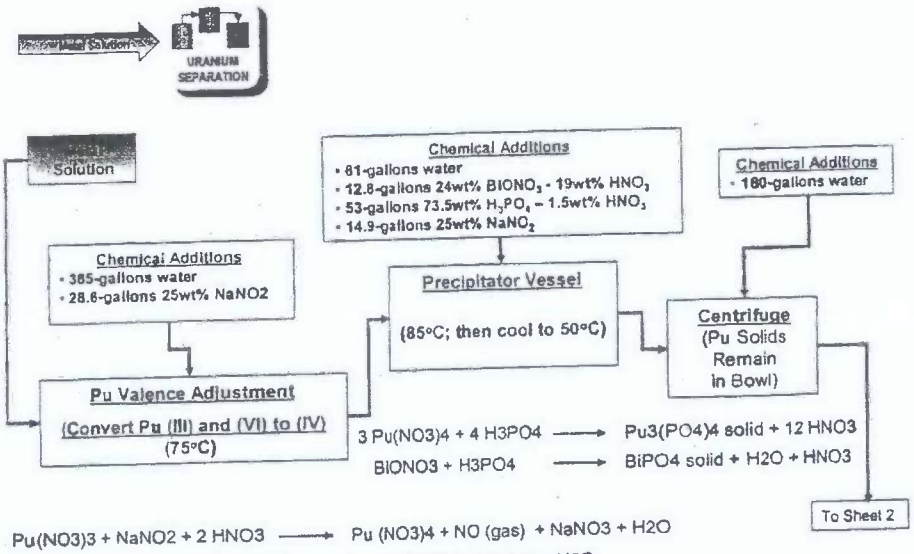
$U + 8\ HNO3 \longrightarrow UO2(NO3)2 + 6\ NO2\ (gas) + 4\ H2O$

$U + 4\ HNO3 \longrightarrow UO2(NO3)2 + 2\ NO\ (gas) + 2\ H2O$

$Pu + 8\ HNO3 \longrightarrow Pu(NO3)4 + 4\ NO2\ (gas) + 4\ H2O$

Note: Pu is dissolved and forms Pu nitrate in the +3, +4, and +6 valence state

Uranium Separation / Plutonium Extraction
(Sheet 1)



$3\ Pu(NO3)4 + 4\ H3PO4 \longrightarrow Pu3(PO4)4\ solid + 12\ HNO3$

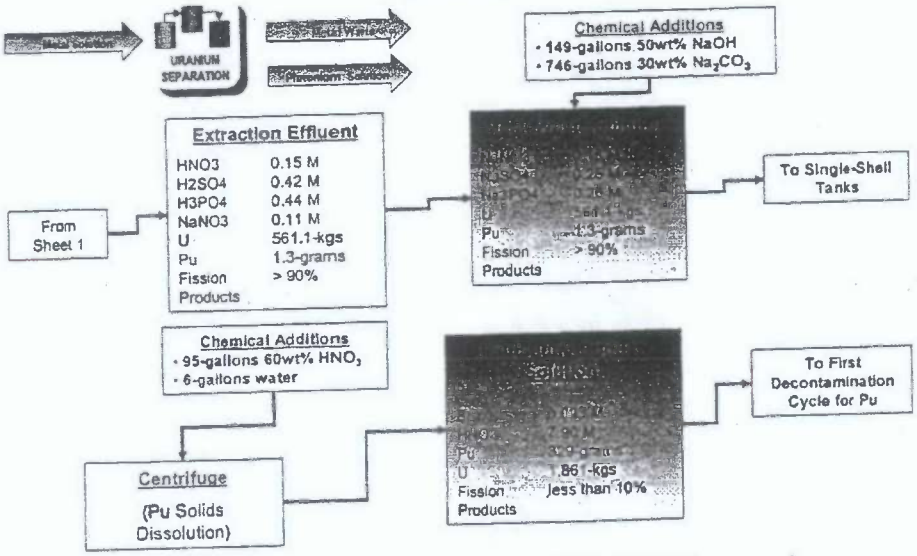
$BiONO3 + H3PO4 \longrightarrow BiPO4\ solid + H2O + HNO3$

$Pu(NO3)3 + NaNO2 + 2\ HNO3 \longrightarrow Pu\ (NO3)4 + NO\ (gas) + NaNO3 + H2O$

$PuO2(NO3)2 + NaNO2 + 2\ HNO3 \longrightarrow Pu(NO3)4 + NaNO3 + H2O$

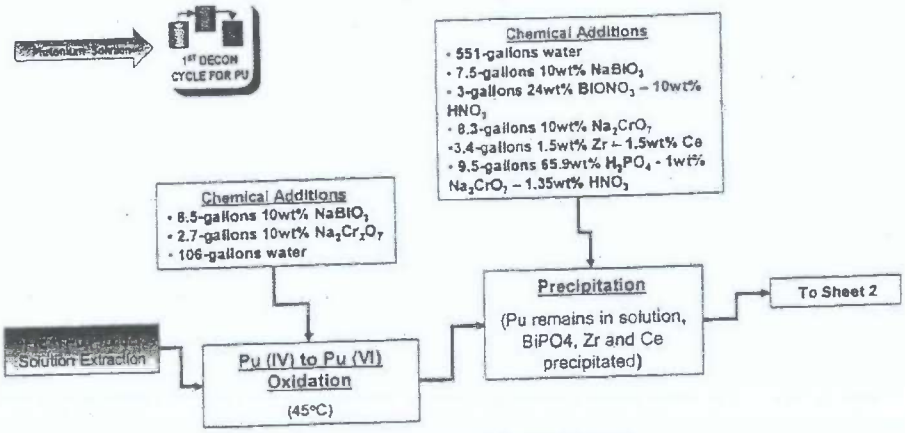
Chemical Reactions for the Bismuth Phosphate Flow Sheet

Uranium Separation / Plutonium Extraction
(Sheet 2)



$Pu_3(PO_4)_4 + BiPO_4 + 15\ HNO_3\ (conc.) \longrightarrow 3Pu(NO_3)_4 + Bi(NO_3)_3 + 5\ H_3PO_4$

First Decontamination Cycle for Plutonium
(Sheet 1)

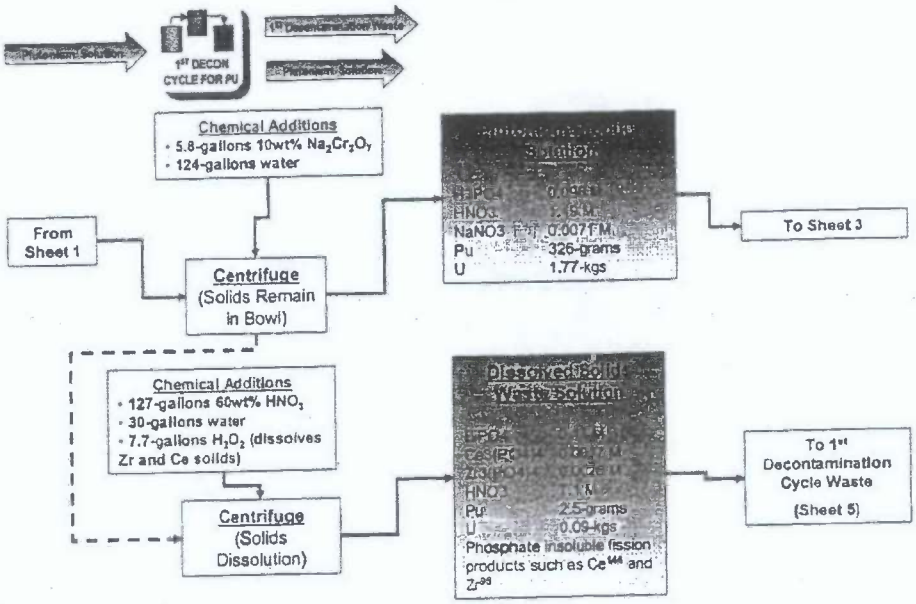


$Pu(NO_3)_4 + NaBiO_3 \longrightarrow PuO_2(NO_3)_2 + BiONO_3 + NaNO_3$

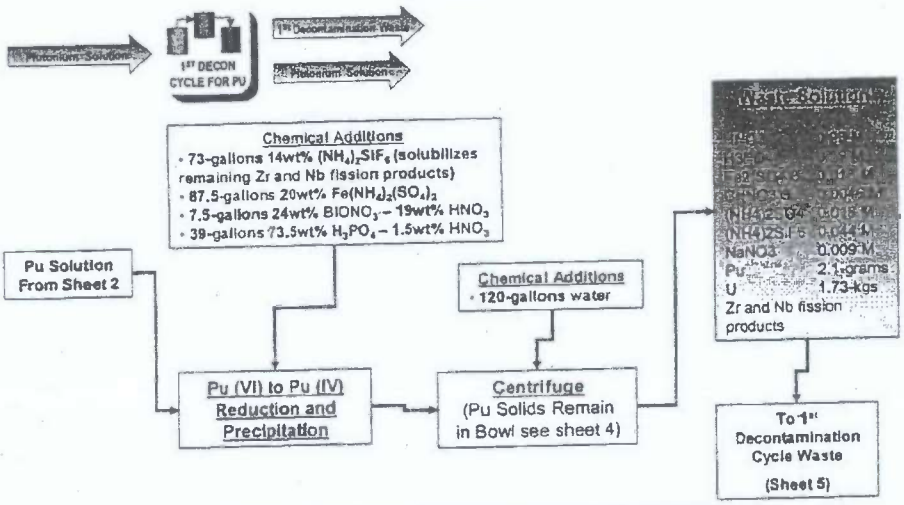
$BiONO_3 + H_3PO_4 \longrightarrow BiPO_4\ (solid) + HNO_3 + H_2O$

Chemical Reactions for the Bismuth Phosphate Flow Sheet

First Decontamination Cycle for Plutonium
(Sheet 2)



First Decontamination Cycle for Plutonium
(Sheet 3)



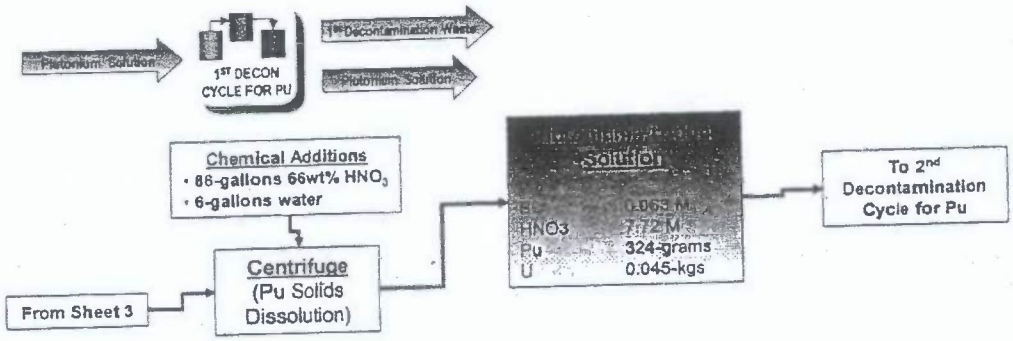
$$PuO_2(NO_3)_2 + 2\,Fe(NH_4)_2(SO_4)_2 + 4\,HNO_3 \longrightarrow Pu(NO_3)_4 + Fe_2(SO_4)_3 + 2\,NH_4NO_3 + (NH_4)_2SO_4$$

$$3\,Pu(NO_3)_4 + 4\,H_3PO_4 \longrightarrow Pu_3(PO_4)_4\ \text{solid} + 12\,HNO_3$$

$$BiONO_3 + H_3PO_4 \longrightarrow BiPO_4\ \text{solid} + H_2O + HNO_3$$

First Decontamination Cycle for Plutonium

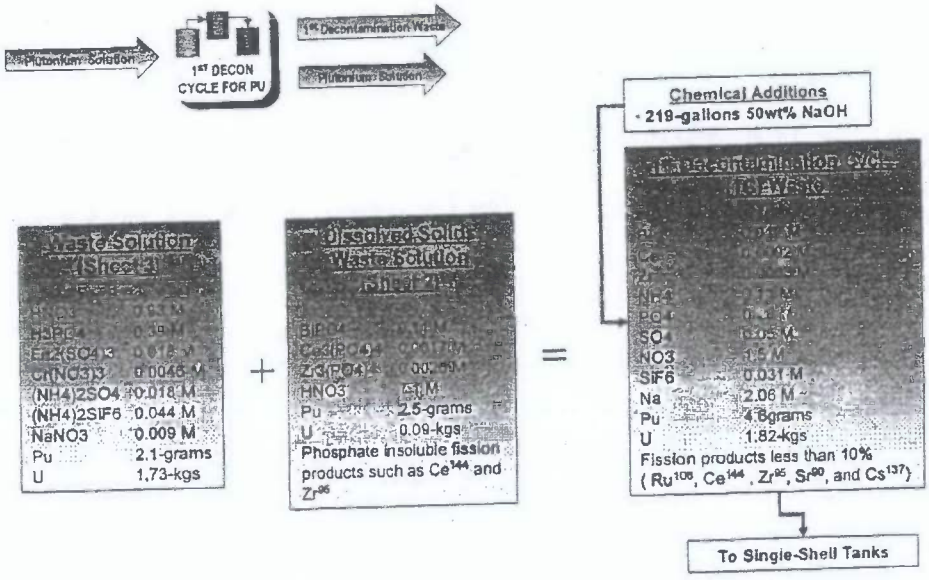
(Sheet 4)



$$Pu_3(PO_4)_4 + BiPO_4 + 15\ HNO_3\ (conc.) \longrightarrow 3Pu(NO_3)_4 + Bi(NO_3)_3 + 5\ H_3PO_4$$

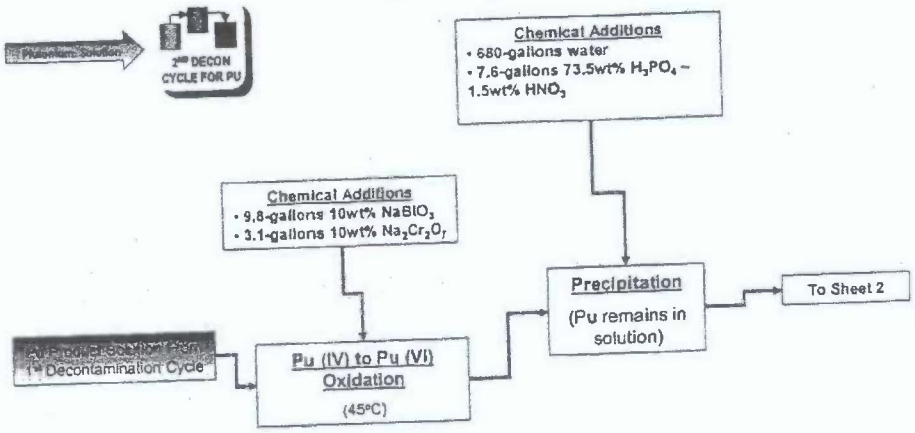
First Decontamination Cycle for Plutonium

(Sheet 5)



Chemical Reactions for the Bismuth Phosphate Flow Sheet

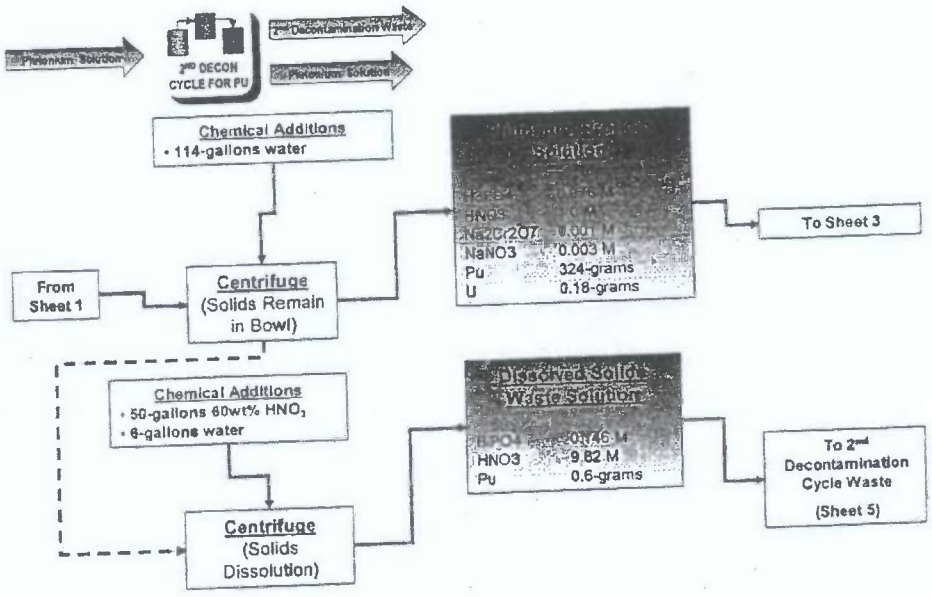
Second Decontamination Cycle for Plutonium
(Sheet 1)



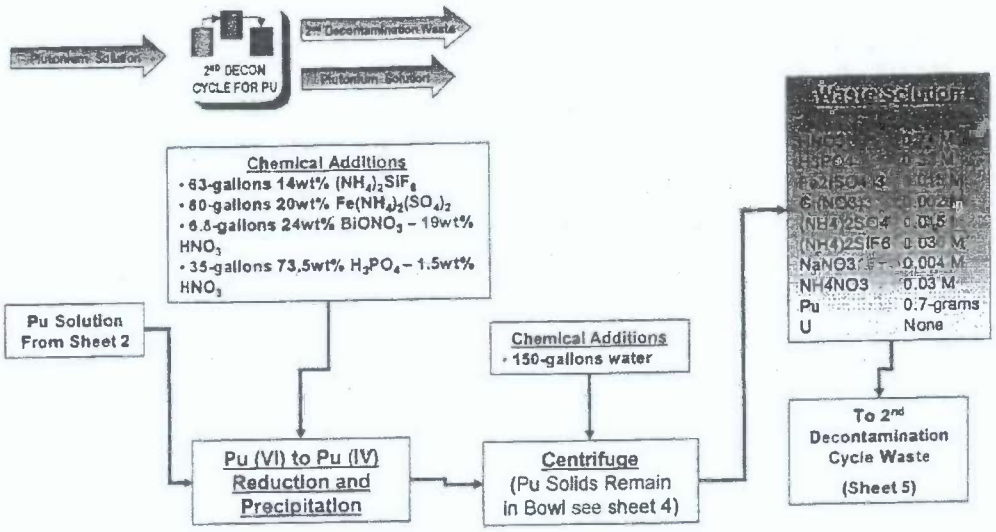
$Pu(NO_3)_4 + NaBiO_3 \longrightarrow PuO_2(NO_3)_2 + BiONO_3 + NaNO_3$

$BiONO_3 + H_3PO_4 \longrightarrow BiPO_4\ (solid) + HNO_3 + H_2O$

Second Decontamination Cycle for Plutonium
(Sheet 2)



Second Decontamination Cycle for Plutonium
(Sheet 3)

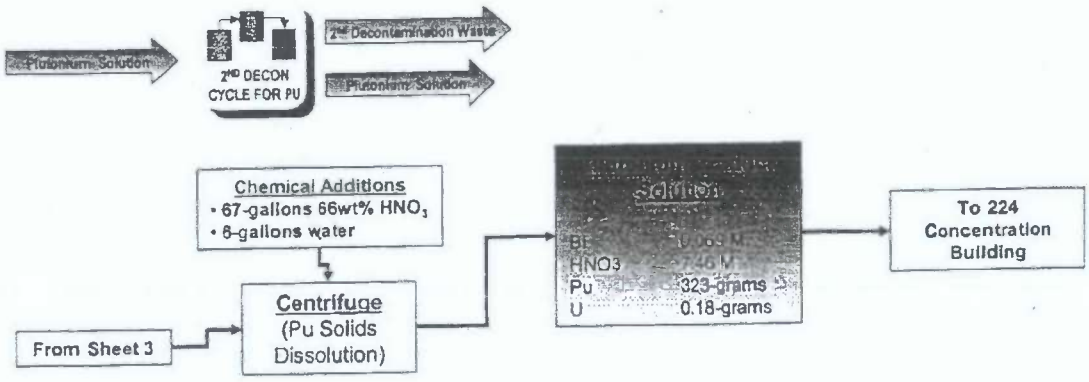


$$PuO_2(NO_3)_2 + 2\,Fe(NH_4)_2(SO_4)_2 + 4\,HNO_3 \longrightarrow Pu(NO_3)_4 + Fe_2(SO_4)_3 + 2\,NH_4NO_3 + (NH_4)_2SO_4$$

$$3\,Pu(NO_3)_4 + 4\,H_3PO_4 \longrightarrow Pu_3(PO_4)_4\ \text{solid} + 12\,HNO_3$$

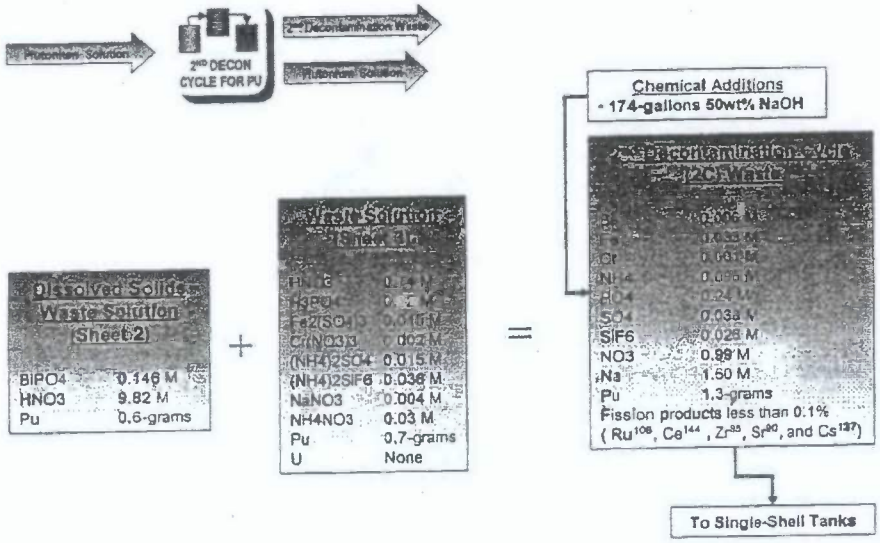
$$BiONO_3 + H_3PO_4 \longrightarrow BiPO_4\ \text{solid} + H_2O + HNO_3$$

Second Decontamination Cycle for Plutonium
(Sheet 4)

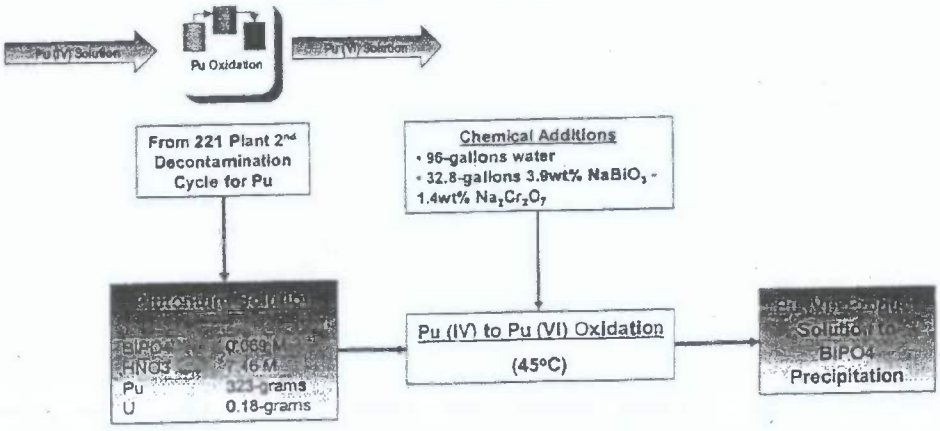


$$Pu_3(PO_4)_4 + BiPO_4 + 15\,HNO_3\ (\text{conc.}) \longrightarrow 3\,Pu(NO_3)_4 + Bi(NO_3)_3 + 5\,H_3PO_4$$

Second Decontamination Cycle for Plutonium
(Sheet 5)

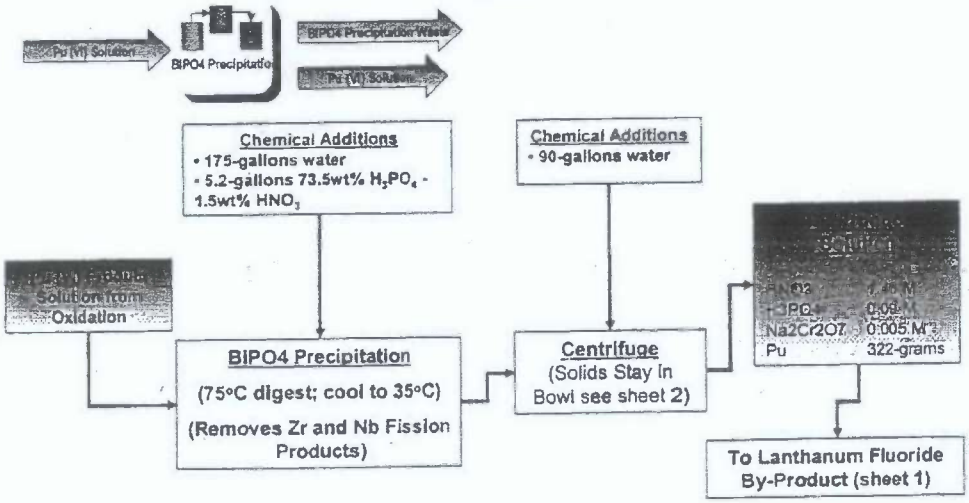


Bismuth Phosphate Cross-Over: Pu (IV) to Pu (VI) Oxidation



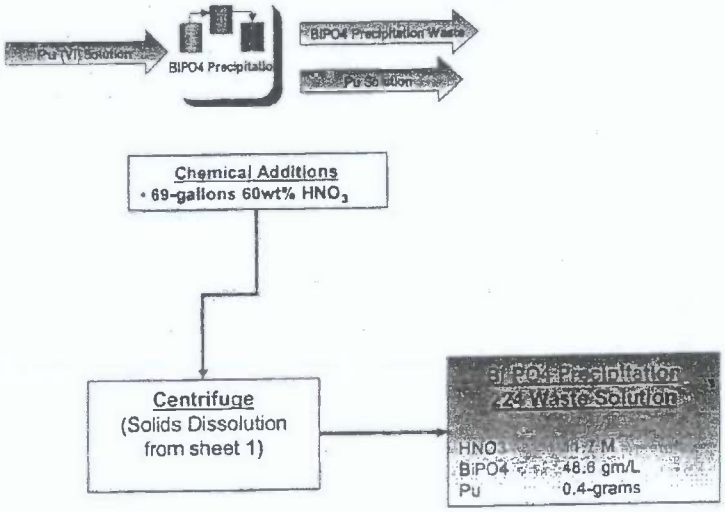
$$Pu(NO_3)_4 + NaBiO_3 \longrightarrow PuO_2(NO_3)_2 + BiONO_3 + NaNO_3$$

Bismuth Phosphate Cross-Over: BiPO4 Precipitation (Sheet 1)

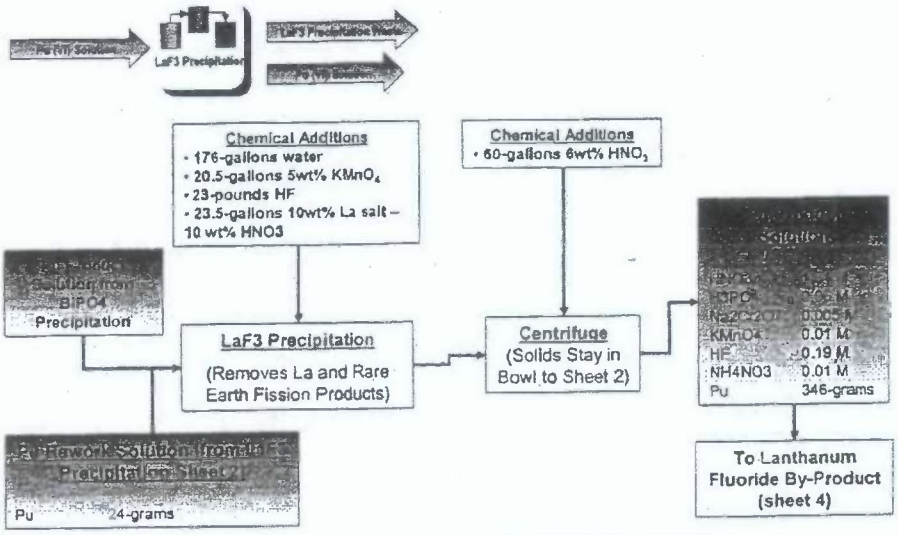


$$BiONO_3 + H_3PO_4 \longrightarrow BiPO_4\ solid + HNO_3 + H_2O$$

Bismuth Phosphate Cross-Over: BiPO4 Precipitation (Sheet 2)

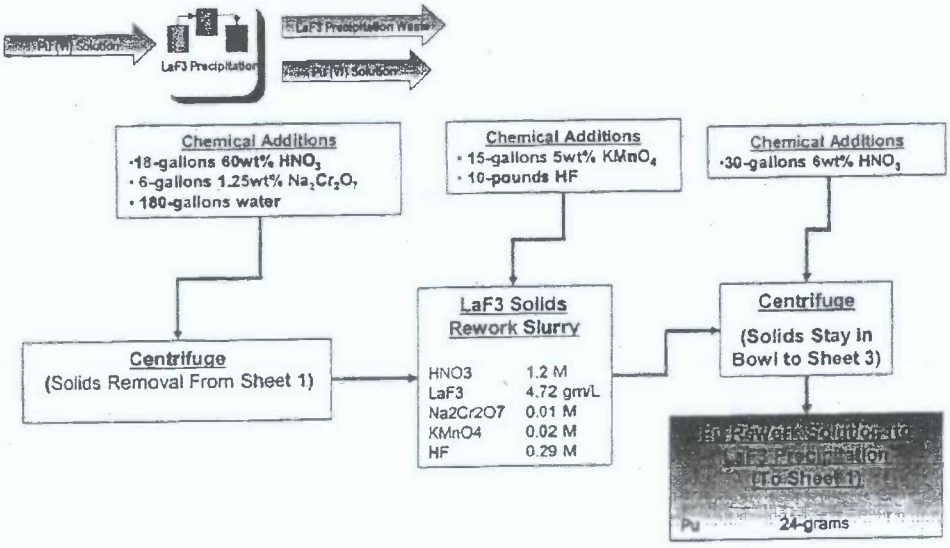


Lanthanum Fluoride By-Product: LaF3 Precipitation (Sheet 1)

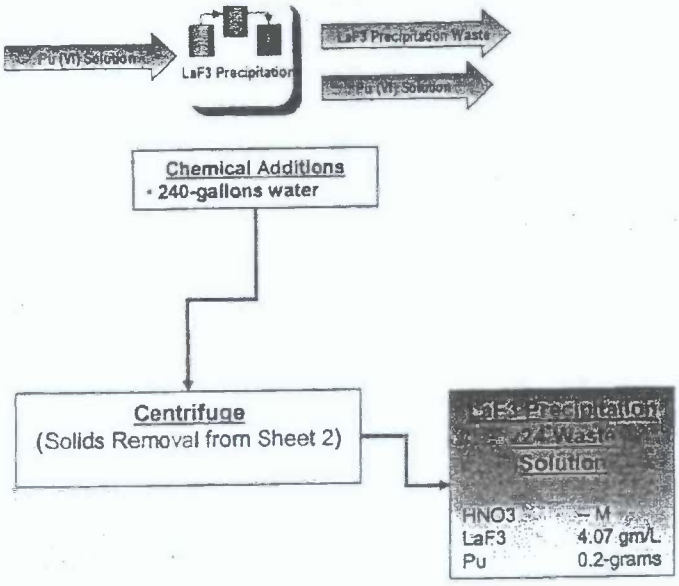


$La(NH_4)_2(NO_3)_5 + 3\ HF \rightarrow LaF_3 + 2NH_4NO_3 + 3\ HNO_3$

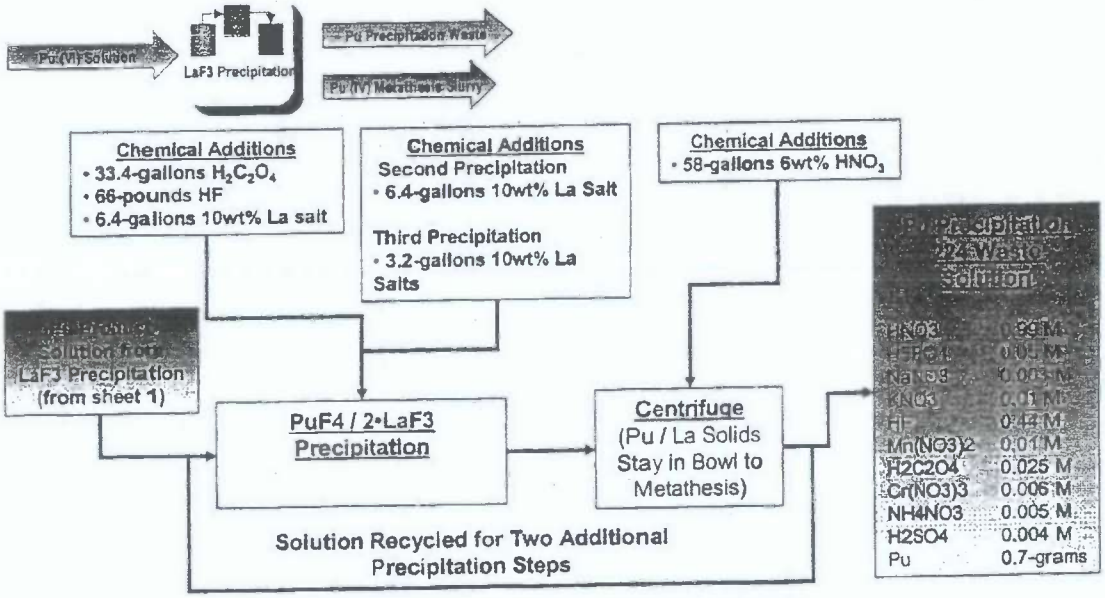
Lanthanum Fluoride By-Product: LaF3 Precipitation (Sheet 2)



Lanthanum Fluoride By-Product: LaF3 Precipitation (Sheet 3)



Lanthanum Fluoride By-Product: Pu (VI) to Pu (IV) Reduction and Precipitation (sheet 4)

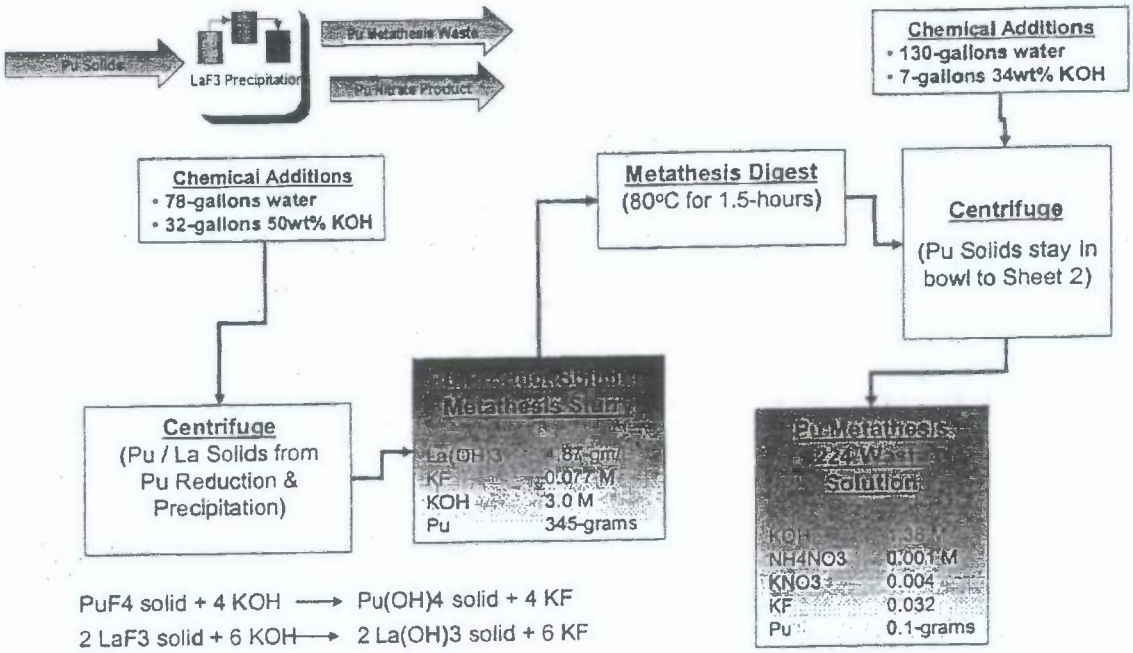


$$2\,HNO_3 + H_2C_2O_4 + PuO_2(NO_3)_2 \longrightarrow Pu(NO_3)_4 + CO_2\,(gas) + 2\,H_2O$$

$$Pu(NO_3)_4 + 4\,HF \longrightarrow PuF_4 + 4\,HNO_3$$

$$2\,La(NH_4)_2(NO_3)_5 + 6\,HF \longrightarrow 2\,LaF_3 + 6\,HNO_3 + 4\,NH_4NO_3$$

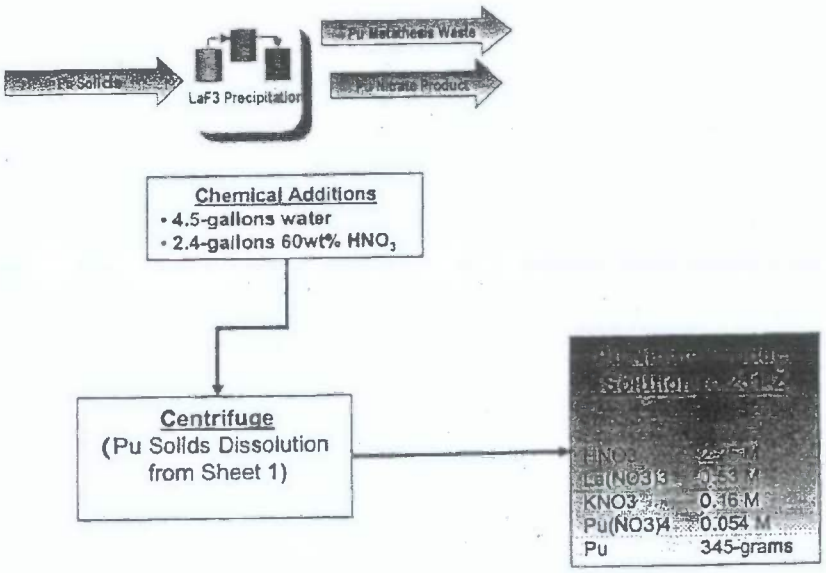
Plutonium Metathesis (Sheet 1)



PuF_4 solid + 4 KOH ⟶ $Pu(OH)_4$ solid + 4 KF

2 LaF_3 solid + 6 KOH ⟶ 2 $La(OH)_3$ solid + 6 KF

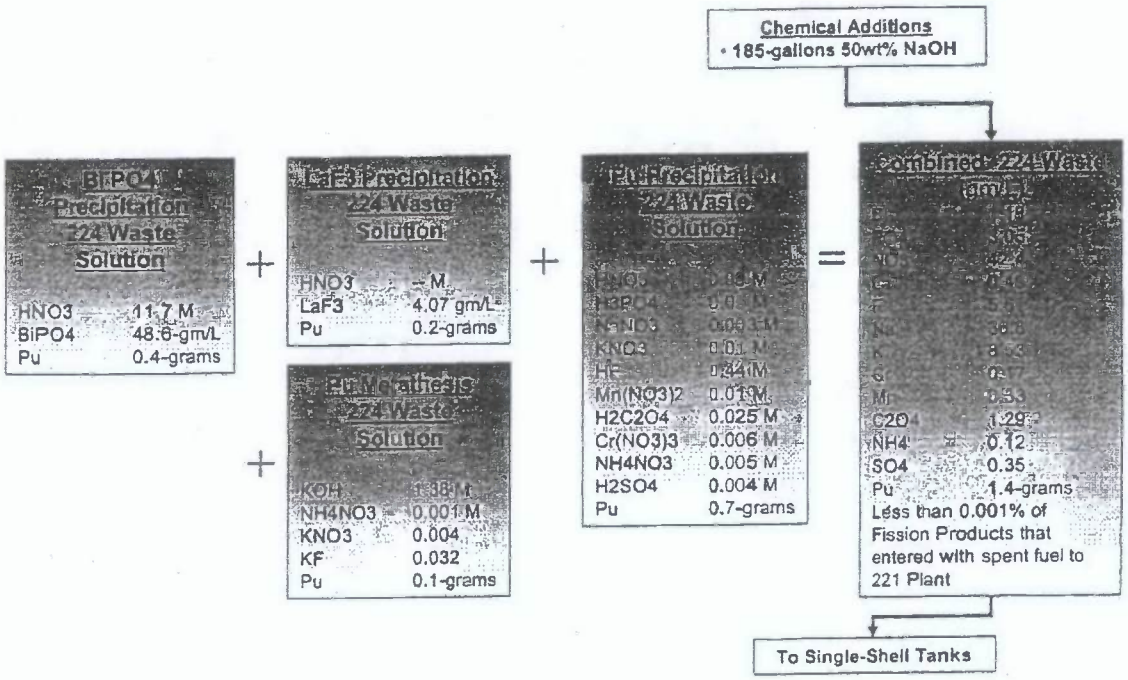
Plutonium Metathesis (Sheet 2)



$Pu(OH)_4$ solid + 4 HNO_3 ⟶ $Pu(NO_3)_4$ + 4 H_2O

$La(OH)_3$ solid + 3 HNO_3 ⟶ $La(NO_3)_3$ + 3 H_2O

224 Building Waste



References for Appendix A Flowsheets:

HW-10475-C, 1944, *Hanford Technical Manual Section C*, General Electric Hanford Atomic Products Operation, Richland, Washington

HW-23043, 1951, *Flow Sheets and Flow Diagrams of Precipitation Separations Process*, General Electric Company, Richland, Washington

HW-26365, 1952, *Brief Summary of Separations Processes*, General Electric Company, Richland, Washington

APPENDIX B – Bismuth Phosphate and PUREX Process Waste Stream Characteristics

Table B-1. Comparison of Bismuth Phosphate and PUREX Process Waste Streams

Waste Stream	Gross Beta Radioactivity µCi/ml	Gross Gamma Radioactivity µCi/ml	Sr-90 µCi/ml	Cs-137 µCi/ml	Waste Batch Volume (gallons)[14]	Waste Batch (gallons / U Ton	Comment
221-T / 221-U Bismuth Phosphate Plant and 224-T / 224-B Pu Concentration Processing Wastes							
Uranium Separations Metal Waste	127	22	Not Reported	Not Reported	2380	3840	Average of three samples taken in 1947. Fission products in Metal waste had decayed for 1 to 2 years; see Table B-2.
Uranium Separations Metal Waste	Not Reported	Not Reported	0.59	57.3	Not Reported		Average Cs and Sr concentrations in Metal waste after U removal in the TBP Plant; see Table B-3
First Pu Product Decontamination Cycle (1C) Waste mixed with Coating Removal Waste (CW)	0.39	0.22	0.0058	0.15	2822	4551	Average fission products concentrations in 1C/CW waste; see Table B-3.
Second Pu Product Decontamination Cycle (2C) Waste	0.0018	0.003	Not Reported	Not Reported	2090	3370	Average fission products concentrations in 2C waste; see Table B-2
224 Pu Product Concentration Building Waste	0.14	0.03	Not Reported	Not Reported	2200	3550	See HW-10728, page 9, 1948, *Process Waste Data – 200 Areas*, Letter from R. S. Bell to file dated August 12, 1948, General Electric Company, Richland Washington
PUREX First Cycle Raffinate							
Waste Stream	**Gross Beta Radioactivity µCi/ml**	**Gross Gamma Radioactivity µCi/ml**	**Sr-90 µCi/ml**	**Cs-137 µCi/ml**	**Waste Batch Volume (gallons)**	**Waste Batch (gallons / U Ton**	**Comment**
PUREX 1WW (concentrated aqueous waste from 1st cycle solvent extraction *before concentration and neutralization; after 1-year decay*)	Not Reported	Not Reported	5,300	5,100	100	100	HW-52824, page 7, 1957, *Ultimate Disposal of PUREX Wastes*, General Electric Company, Richland Washington. (1957 PUREX Flowsheet for processing 600 MWD / ton natural U fuel)
PUREX 1WW (concentrated aqueous waste from 1st cycle solvent extraction *after sugar denitration, concentration and neutralization*)	Not Reported	Not Reported	~ 218,800 (includes Sr^{89} and Sr^{90})	~12,700	657.4 (assuming 16-tons Uranium processed per batch)	41.1	ARH-214, 1968, PUREX Chemical Flowsheet Processing Aluminum Clad Uranium Fuels, Atlantic Richfield Hanford Company, Richland Washington. (1968 PUREX Flowsheet for processing 600 MWD/ton natural U fuel; includes internal recycle of wastes and sugar denitration of 1WW)

[14] Bismuth Phosphate Process waste volumes are from HW-23043, 1951, *Flow Sheets and Flow Diagrams of Precipitation Separations Process*, General Electric Company, Richland, Washington

Table B-2. Analyses of Bismuth Phosphate Process Supernatant

Waste Type[1,2]	Tank	Date Filled	pH	Pu Gm/liter	Gross Beta millicuries/liter	Gross Gamma millicuries/liter	Date Sampled
Metal Waste	T-101		10.1	70	200[5]	70[5]	12-12-1946
Metal Waste	T-101	08/1945	10	35	110	25	7-01-1947
Metal Waste	T-102	11/1945	9.9	60	120	20	7-01-1947
Metal Waste	T-103	02/1946	9.8	60	150	20	7-01-1947
	Average for three samples taken in 1947			51.7	126.7	21.7	
1C/CW	B-109	04/1946	9.9	40	0.65	0.28	3-18-1947
1C/CW	C-112	04/1947	9.9	12	12	4.4	3-18-1947
2C[4]	B-111	04/1946	6.9	7.2E-02	2.0E-03	3.0E-03	7-1-1947
2C	B-112	08/1946	6.8	4.32E??[3]	1.5E-03	3.0E-03	7-1-1947

Notes:

(1) See HW-10728, 1948, *Process Waste Data – 200 Areas*, Letter from R. S. Bell to file dated August 12, 1948, General Electric Company, Richland Washington and HW-3-3220, 1945, *A Study of Decontamination Cycle Waste Solutions and Methods of Preparing Them for Disposal*, E. I. Du Pont De Nemours Company, Richland Washington.

(2) Solids formed in each of wastes, settling to the bottom of each tanks. These sample analyses are for the supernatant only and are not representative of the sludges.

(3) The reported Pu sample analyses for tank B-112 seems to be in error and lacking an exponent in HW-10728.

(4) Prior to October 1945, the 2C waste was neutralized to a pH of approximately 10. The waste collected in tanks 241-T-110, 241-T-111, and 241-T-112 were neutralized to about pH 7 after October 1945 to precipitate bismuth and plutonium (HW-3-3220, page 13).

(5) Reduction in the gross gamma and beta analyses for the metal waste in tank T-101 from sampling in 12-12-1946 to 07-01-1947 is due to decay of short-lived fission products.

Table B-3. Radionuclide Analyses of Metal Waste and First Decontamination Cycle Waste

Tank	Date Filled	Pu µgm/cc	Gross Beta µCi/cc	Gross Gamma µCi/cc	Sr µCi/cc	Cs µCi/cc	Ru µCi/cc	Rare Earths – Y – Ce µCi/cc	Ce µCi/cc	Nb µCi/cc	Zr µCi/cc	Te µCi/cc
Analyses of Metal Waste Supernatant Following Uranium Extraction [1]												
C-106	Not specified				0.44	54.2						
BX-108	Not specified				0.26	132.4						
BX-109	Not specified				1.08	56.3						
C-112	Not specified				1.20	25.8						
C-109	Not specified				0.46	40.7						
C-111	Not specified				0.10	34.5						
Average Concentrations for Metal Waste					**0.59**	**57.3**						
Analyses of First Decontamination Cycle Waste Mixed with Coating Removal Waste Supernatant [2]												
B-107	8-1945	1.7E-02	0.135	0.055	0.011	0.10						
T-107	9-1945	1.5E-03	0.170	0.093	0.0013	0.20						
B-108	12-1945	2.0E-02	0.183	0.044	0.022	0.12						
T-108 (Top)	12-1945	2.0E-02	0.25	0.073	0.012	0.17	0.0066	0.047	0.007	0.0018	0	1.2E-05
T-108 (Bottom)	12-1945	2.0E-02	0.25	0.070	0.012		0.0065	0.029	0.0066	0.0024	0	3E-05
T-109	3-1946	2.6E-03	0.14	0.082	0.00038	0.15						
B-109	4-1946	1.8E-02	0.16	0.051	0.01	0.11						
T-104 (Top)	7-1946	3E-03	0.51	0.130	0.00013	0.13	0.058	0.004	0.051	0.028	0.010	2.4E-05
T-104 (Bottom)	7-1946	3E-03	0.52	0.160	0.00037		0.059	0.003	0.050	0.028	0.015	3.6E-05
C-110	8-1946	2E-03	0.14	0.0067	0.00026	0.11						
C-111	11-1946	4.2E-03	0.16	0.069	0.01	0.13						
C-112	4-1947	3.1E-03	0.14	0.064	0.006	0.13						
U-110	4-1947	2.1E-04	0.13	0.069	0.00011	0.17						
U-111	10-1947	3.4E-04	0.12	0.060	0.00023	0.14						
TX-109 [3]	9-1949	2.7E-05	2.8	2.2	0.00087	0.27	0.34	0.0085	0.0035	0.34	1.2	8E-05
Average Concentrations for 1C / CW		**7.67E-03**	**0.39**	**0.22**	**0.0058**	**0.15**						

Notes:

[1] HW-36717, Decontamination of Uranium Recovery Process Stored Wastes Interim Report, May 16, 1955, W. W. Schulz, General Electric Company, Richland Washington.

[2] HW-20195, Radioactive Content of Stored Bismuth Phosphate First Cycle Waste Supernatants, February 5, 1951, General Electric Company, Richland Washington.

[3] Tank TX-109 exhibits higher gross beta and gross gamma radioactivity since this tank was sampled shortly after filling and the short-lived fission products (e.g., Ru, Nb, and Zr) had not decayed appreciably.

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Transuranic Waste Project Management Plan

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Richland, WA 99352
U.S. Department of Energy Contract DE-AC27-99RL14047

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Key Words: TRU, Transuranic, Plan, Waste Packaging, Waste Retrieval

Abstract: This document describes the overall strategy, objectives, and contractor management requirements that define how CH2M HILL Hanford Group Inc. will execute the Transuranic (TRU) Waste Project.

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TRANSURANIC WASTE PROJECT MANAGEMENT PLAN

September 30, 2003

Prepared By
CH2M HILL Hanford Group, Inc.
Richland, Washington

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LIST OF TERMS

BCR	baseline change request
CCP	Centralized Characterization Project
CH2M HILL	CH2M HILL Hanford Group, Inc.
CSS	Characterization, Storage, and Shipping
DOE	U.S. Department of Energy
ES&H	Environment, Safety, and Health
ICD	interface control document
ISF	interim storage facility
ORP	Office of River Protection
PMP	project management plan
RCRA	*Resource Conservation and Recovery Act of 1976*
TFC	Tank Farm Contractor
TRU	transuranic
TRUPACT-II	Transuranic Packaging Transport, Model II
WBS	work breakdown structure
WIPP	Waste Isolation Pilot Plant
WPS	waste packaging system
WRS	waste retrieval system
WTP	Waste Treatment Plant

1.0 INTRODUCTION

This project management plan (PMP) documents the overall strategy, objectives, and contractor management requirements that define how CH2M HILL Hanford Group, Inc. (CH2M HILL) will execute the Transuranic (TRU) Waste Project. The project objective is to treat and disposition TRU tank wastes:

- Identification of candidate tanks storing TRU waste
- Retrieval of TRU wastes from single-shell tanks
- Packaging of the TRU wastes into containers compliant with Waste Isolation Pilot Plant (WIPP) requirements
- Disposition of secondary wastes.

The project is also responsible to plan for and ensure the following government-furnished services and items are integrated:

- Interim storage of packaged TRU wastes
- Waste characterization
- WIPP certification of packaged TRU wastes
- Preparation and loadout of WIPP certified packages for transportation to WIPP.

This PMP establishes the project's roles, responsibilities, goals, and objectives for contact-handled TRU waste. This PMP will be updated as required to reflect changes as the project moves through the design, fabrication, testing, and installation phases and as required to address remote-handled TRU waste.

1.1 BACKGROUND

The strategy presented in *Performance Plan for Accelerated Cleanup of the Hanford Site* (DOE/RL-2002-47) incorporates a combination of increased high-level waste vitrification capacity, increased low-activity waste output, and the use of supplemental technology to augment the Waste Treatment Plant (WTP) processing capacity. The successful deployment of cost-effective non-WTP supplemental treatment techniques will reduce the double-shell tank space usage, will reduce the planned loading on the WTP, and will help ensure that the treatment of Hanford tank wastes will be completed by 2028. The strategy of interest in this PMP is the deployment of technologies to package TRU tank wastes for final disposition at WIPP.

1.2 PROJECT OBJECTIVES

Remove and treat 1 million gallons (in tank volume) by September 30, 2006 of either TRU sludge or other low-level tank waste. A minimum of 750,000 gallons of TRU waste shall be included. Out-of-scope but related objectives include characterization, certification, interim storage, and shipment to disposal.

Turnover of retrieved tanks to the closure project for interim closure of additional single-shell tanks by September 30, 2005.

Implementation of this project will result in the following endpoint/outcome:

- A net savings relative to the U.S. Department of Energy (DOE) established costs of treating the wastes via vitrification in the WTP
- All retrieved tank waste completes treatment by 2028 via a combination of supplemental non-WTP treatment and treatment in the WTP
- Wastes with high sulfate, chromate, or other chemical constituents that limit overall waste loading in glass are preferentially treated outside the WTP, resulting in higher net WTP throughputs and efficiencies.

1.3 SCHEDULE OBJECTIVES

Table 1 lists the scheduled activities that meet the objectives of the TRU Waste Project.

Table 1. Transuranic Waste Project Schedule Objectives

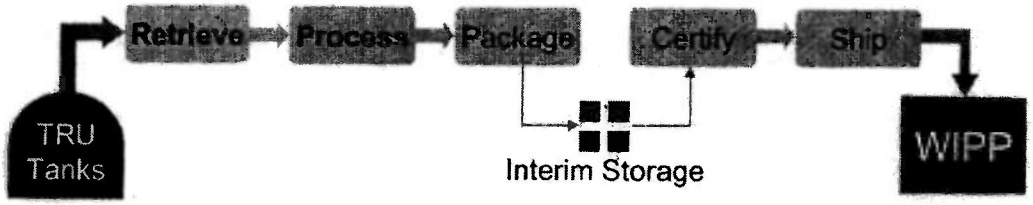
Objective	Date
Initiate retrieval of TRU waste from single-shell tanks (IMES) Initiate treatment and packaging of TRU retrieved from single-shell tanks (IMES Milestone)	10/28/04
Initiate shipment of packaged TRU waste to WIPP (Gold Chart)	6/28/05
Turnover retrieved tanks to closure project for interim closure of additional single-shell tanks	9/30/05
Treat and disposition as non high-level waste, 1 million gallons of tank waste that is either TRU sludge or other low-level tank farm waste, of which a minimum of 750,000 gallons is TRU waste (PBI 4).	9/30/06

IMES = integrated mission execution schedule.
PBI = performance-based incentive.
TRU = transuranic.
WIPP = Waste Isolation Pilot Plant.

1.4 PROJECT DESCRIPTION

The TRU Waste Project encompasses identification of candidate tanks, waste retrieval, waste processing, treatment, and handling. A block flow diagram of the project is shown in Figure 1. A detailed integrated flowsheet will be developed as the project matures. The retrieved tank waste will be transferred via overground hose-in-hose transfer lines to the TRU waste packaging system (WPS) where the waste is treated and packaged. The packaged TRU waste will be placed in interim on-site storage pending WIPP certification. Final disposition of TRU waste is at WIPP.

Figure 1. Transuranic Waste Project Block Flow Diagram



1.4.1 Transuranic Waste Retrieval

The Retrieval Project Manager is responsible for the retrieval portion of the TRU Waste Project scope. Eight 200-series tanks and one 100-series tank have been identified as containing TRU wastes. These nine tanks and their associated volumes of wastes are shown in Table 2.
In addition, other tank waste may qualify for disposition by the TRU Waste Project. Waste in the following tanks is currently being evaluated: B-107, B-110, B-111, T-104, T-105, T-106, T-107, T-110, and T-112.

Table 2. Tanks Containing Transuranic Waste and Waste Volumes

Tank Number	Tank Integrity	Sludge (kgal)	Supernate Liquid (kgal)	Total Waste (kgal)
B-201	Assumed Leaker	30	0	30
B-202	Sound	29	0	29
B-203	Assumed Leaker	51	1	52
B-204	Assumed Leaker	50	1	51
T-111	Assumed Leaker	447	0	447
T-201	Sound	29	2	31
T-202	Sound	21	0	21
T-203	Sound	37	0	37
T-204	Sound	37	0	37
	Total	731	4	735

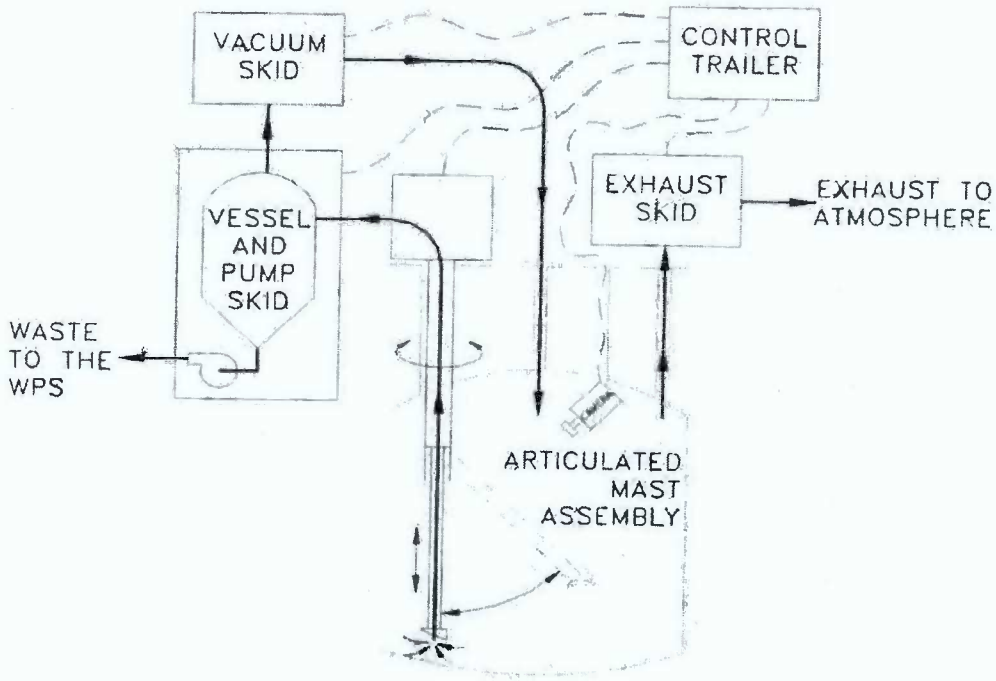
Source: Mackay 2002.

The temporary staging of T-200-series tank waste in T-100-series tanks is being evaluated. If this approach is taken, waste would be transferred from the T-200-series tanks to one of the T-100-series tanks for staging until the packaging effort commences. All of the staged waste would then be retrieved to packaging. The retrieval sequence will be documented in the tank waste retrieval interface control document (ICD).

1.4.1.1 200-Series Tank Waste Retrieval System

The 200-Series Tanks Waste Retrieval Project will use a vacuum retrieval system that utilizes an articulating mast to remove waste. The system is designed to loosen, scarify or otherwise dislodge waste as necessary to facilitate the vacuum process. The waste is first accumulated in a batch tank, and then transferred to the waste receiver using hose-in-hose transfer lines. For the B-200-series tanks, the waste receiver will be the WPS. In the case of the T-200-series tanks, the waste receiver will either be tank T-110, tank T-111, or the WPS (Figure 2), depending on the waste volume in tank T-110 or T-111 at the time of waste retrieval.

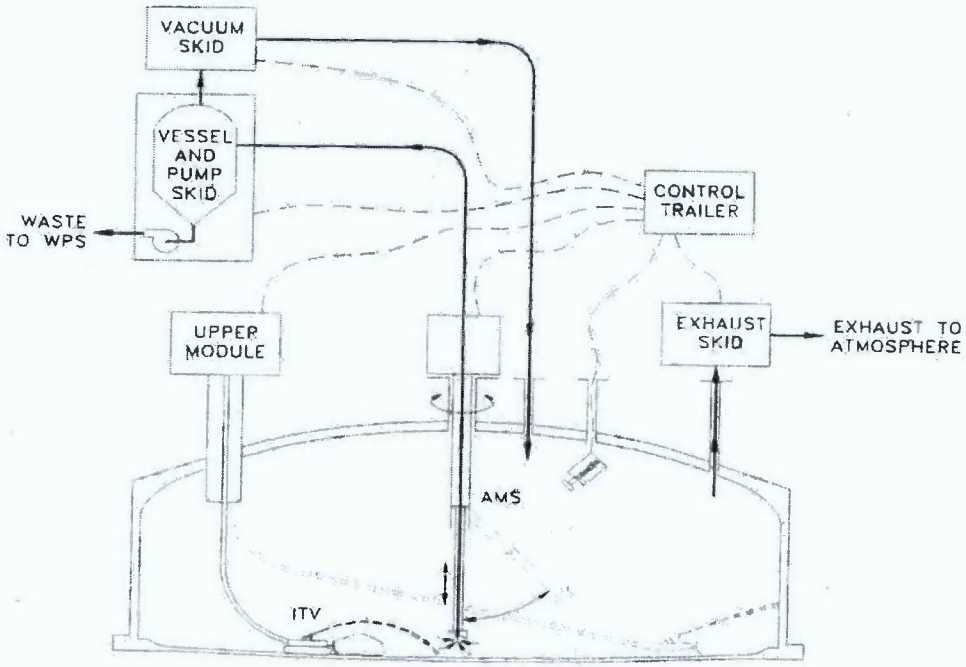
Figure 2. 200-Series-Tank Waste Retrieval System Diagram



1.4.1.2 100-Series Tank Waste Retrieval System

The Tank T-111 Waste Retrieval Project will use the same technology and process as described for the T-200-series tanks. As tank T-111 has a larger diameter than the 200-series-tanks, the articulating mast is augmented by the use of an in tank vehicle that will facilitate the movement of waste to the vacuum system. The waste is transferred to the WPS using hose-in-hose transfer lines (Figure 3).

Figure 3. 100-Series Tank Waste Retrieval System Diagram



1.4.2 Transuranic Waste Packaging System

The Packaging Project Manager is responsible for the packaging portion of the TRU Waste Project scope. The WPS receives material from the waste retrieval system (WRS) and packages it in containers suitable for shipment to an offsite geologic repository. The TRU waste from the WRS will be dewatered to optimize container loading and placed in a standard waste box, per WIPP waste acceptance criteria, for eventual shipment to the WIPP. The WPS provides the following main functions:

- **Waste Handling** – Receives the waste stream from the WRS, provides lag storage for processing, and samples/characterizes the waste per quality assurance protocols
- **Package** – Places the waste in containers suitable for WIPP shipment
- **Water Management** – Dewaters the incoming waste stream, provides a recycle stream to the WRS, disposes of the excess effluent, optimizes water content of the waste package, and adds sorbent to the waste package.

The WPS will be built as modules for simple relocation between tank farms. The WPS will be self-contained by providing all necessary support systems including: power, water, ventilation, control room, and personnel interface.

Up to two WPS units will be fabricated to maximize efficient interface at the two tank farms containing contact-handled TRU waste (B and T tank farms), depending upon retrieval rates and schedules.

1.4.3 Transuranic Interim Storage

For the storage portion of the TRU Waste Project scope, the Characterization, Storage, and Shipping (CSS) Project Manager is responsible to plan for and ensure the following government furnished services and items are integrated with the project.

The WIPP Certification process requires head space gas sampling of each TRU waste container after a specified storage period (referred to as the drum age criteria). Before head space gas sampling, the container is required to equilibrate at a minimum temperature of 65 °F for 72 hours. The interim storage facility (ISF) will need to provide *Resource Conservation and Recovery Act of 1976* (RCRA)-permitted waste storage for up to 1,500 standard waste boxes for greater than 90 days. Head space gas sampling capability may be provided in the ISF or in a separate facility.

The ISF options under evaluation include utilizing the existing site waste storage facilities and/or constructing soft sided, tent-like structures to be used in combination with other existing onsite structures (e.g., B-616) to support head space gas sampling temperature equilibration and potentially load management. ISF selection will be based on the implementation cost and risk associated with each option.

Before placing the containers in storage, a non-destructive assay will be performed. A RCRA 90-day storage pad may be required to provide operational flexibility of the non-destructive assay system by providing a staging area for the waste containers before non-destructive assay is performed.

1.4.4 Transuranic Waste Shipping to Waste Isolation Pilot Plant

For the shipping portion of the TRU Waste Project scope, the CSS Project Manager is responsible to plan for and ensure the following government furnished services and items are integrated with the project.

WIPP will provide the transportation system for transporting the TRU waste to WIPP, including the Transuranic Packaging Transport, Model II (TRUPACT-II) containers, truck/trailer, and truck drivers. The TRU Waste Project is responsible for preparing the waste payloads, preparing the TRUPACT-IIs for shipment, loading payloads into the TRUPACT-IIs, and providing the transportation documentation in accordance with the WIPP and TRUPACT-II requirements. The contact-handled TRU waste shipping profile is planned in the TRU Waste National Shipping Schedule as follows:

- 1 shipment in third quarter of 2005
- 1 shipment per week from the fourth quarter of 2005 through the fourth quarter of 2006
- 8 shipments per week from the first quarter of 2007 through the third quarter of 2008.

TRUPACT-IIs loading and testing requires a highly specialized facility capable of disassembly, maintenance, loading and leak testing of TRUPACT-IIs. Options currently under evaluation include: use existing site infrastructure; WIPP Centralized Characterization Project (CCP) provided mobile loading systems; constructing stand alone facility for WIPP loading.

1.4.5 Transuranic Waste Isolation Pilot Plant Waste Program

For the shipping portion of the TRU Waste Project scope, the CSS Project Manager is responsible to plan for and ensure the following government furnished services and items are integrated with the project.

A WIPP TRU Waste Program will be developed to translate WIPP requirements to specific operational and design requirements. The waste certification document hierarchy is shown in Figure 4. Implementation of WIPP requirements will be facilitated by the use of the WIPP CCP. The CCP will provide all WIPP-specific programmatic documents as well as performing WIPP-required waste characterization under a WIPP certified program. A statement of work and an interface document are being developed between CH2M HILL and the CCP to define work requirements and responsibilities for providing materials, equipment and services. An Acceptable Knowledge Program will be developed to support compliance with the WIPP TRU Waste Program. Waste characterization required by WIPP will be completed by utilizing the WIPP CCP resources in conjunction with other site resources and the Retrieval and Packaging Projects. Strict compliance with WIPP requirements will be enforced from characterization through storage and shipment. Waste must be managed and the acceptable knowledge documented from the point of generation through disposal at the WIPP site.

Upon WIPP certification of the TRU waste and approval of the TRUPACT-II payload assembly, the standard waste boxes will be loaded into the WIPP provided TRUPACT-IIs for shipment and disposal of waste at WIPP.

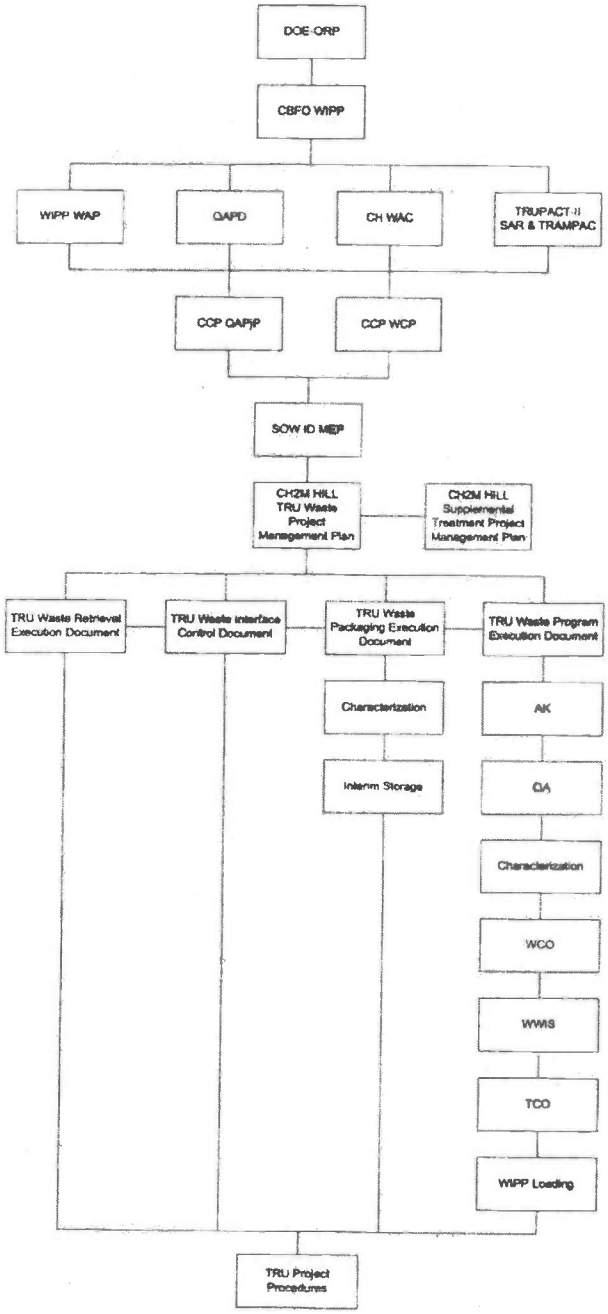
1.4.6 Secondary Waste

For the disposition of secondary wastes from execution of the TRU Waste Project scope, the CSS Project Manager is responsible to plan for and ensure the following government furnished services and items are integrated with the project.

The Waste Packaging Project will generate liquid effluent that will need to be transported to the Effluent Treatment Facility for treatment and disposal. The Waste Packaging Project will provide a RCRA-compliant tanker filling station. Transportation treatment and disposal of the liquid effluent is currently anticipated to be provided by the Effluent Treatment Facility but will need to be confirmed and documented in an ICD.

Other miscellaneous secondary wastes will be packaged in approved containers by the generating project that will be dispositioned through coordinated management with the solid waste disposal organization, including receipt, transportation, treatment, and disposal.

Figure 4. Transuranic Waste Certification Document Hierarchy



AK = acceptable knowledge.
CBFO = Carlsbad Field Office.
CCP = Central Characterization Project.
DOE = U.S. Department of Energy.
ID = interface document.
MEP = mobilization execution plan.
ORP = Office of River Protection.
QA = quality assurance.
QAPD = Quality Assurance Program document.
QAPjP = CCP Transuranic Waste Characterization Quality Assurance Project Plan.
SAR = safety analysis report.
SOW = statement of work.
TCO = Transportation Certification Official.
TRAMPAC = TRUPACT-II Authorized Methods for Payload Control.
TRU = transuranic.
TRUPACT-II = Transuranic Packaging Transport, Model II.
WAC = waste acceptance criteria.
WAP = waste analysis plan.
WCO = Waste Certification Official.
WCP = waste certification plan.
WIPP = Waste Isolation Pilot Plant.
WWIS = WIPP Waste Information System.

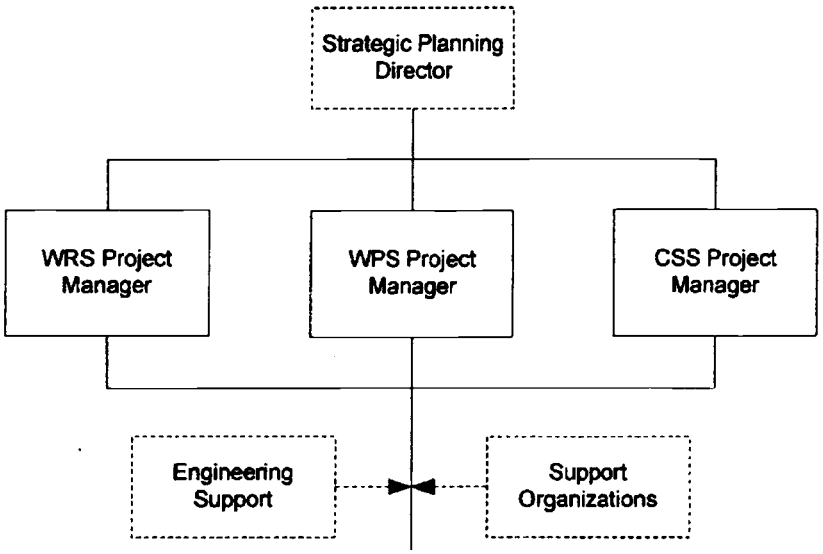
2.0 MANAGEMENT STRUCTURE

This section focuses on defining the organizational structure, relationships, roles, and responsibilities of CH2M HILL relevant to the TRU Waste Project. It also addresses the Office of River Protection (ORP) role in overseeing the project.

2.1 PROJECT STRUCTURE

The organizational structure for this project is shown in Figure 5. The corresponding participant roles and responsibilities are defined in the following sections.

Figure 5. Transuranic Waste Project Organization



CSS = characterization, storage, and shipping.
TRU = transuranic.
WRS = waste retrieval system.
WPS = waste packaging system.

2.2 ROLES AND RESPONSIBILITIES

2.2.1 U.S. Department of Energy, Office of River Protection

The ORP role is to provide contractual direction to CH2M HILL via the ORP Contracting Officer and designated Contracting Officer Representatives. Responsibilities include:

- Provides interface coordination with other DOE offices.
- Provide technical and contractual direction to CH2M HILL.
- Provide technical oversight of CH2M HILL to ensure the products and services for which the Contracting Officer or Contracting Officer Representative is responsible are delivered in accordance with the contract terms and conditions.

- Review invoices, as required.
- Provide approval of multi-year work plan documentation.

2.2.2 CH2M HILL Hanford Group, Inc.

2.2.2.1 Transuranic Waste Project

The TRU Waste Project is organized to take advantage of the corporate structure of CH2M HILL. The project will operate as a matrixed project and will not consist of a full-time dedicated staff.

The three elements of the project are (1) Retrieval Project, (2) Packaging Project, and (3) CSS Project. All three projects must be sensitive to the highly compliant nature of the end state, producing a WIPP-compliant package. Seemingly innocuous changes in approach, chemical additions, process, or paperwork can have a significant negative impact.

The Retrieval Project is responsible for all activities required to design, procure, install, and operate systems that retrieve waste from the tanks and provide it to the packaging project. The Packaging Project is responsible for receiving the waste from the retrieval project and packaging the waste for eventual shipment to the WIPP. The CSS Project is responsible for integrating the three projects and establishing options for the implementation of the government furnished services and items that include interim storage facilities, characterization and certification activities, and facilities for transportation loading for eventual shipment to WIPP.

The CSS Project is focused across the TRU Waste Project and externally. The Retrieval and Packaging Projects are largely focused within the company.

The Strategic Planning Director is responsible for ensuring the project scope is defined, scheduled developed, execution plans established, and that the project is transitioned to the field for execution.

The Strategic Planning Director maintains oversite of the project to ensure the project will reach the desired outcome and remains consistent with the integrated mission objectives. The strategic planning director is the primary interface with the client, WIPP, regulators, and stakeholders.

2.2.2.1.1 Retrieval, Packaging, and Characterization, Storage and Shipping Project Managers

The TRU Waste Project Managers provide day-to-day management and oversight of their project activities, including interfaces with ORP, subcontractors, and CH2M HILL organizations (e.g., Program Office, Tank Farm Operations); approval/administration of project baselines; management of project cost account plans; technical direction to the subcontractors; coordination of risk identification/analysis/mitigation activities; support for budget validations and major reviews; and Operations Liaison support between project team and Tank Farm Operations functions during design/installation phases.

2.2.2.1.2 Engineering

The Tank Farm Contractor (TFC) engineering organization as, outlined in *Engineering Program Management Plan* (TFC-PLN-03), manages information defining the requirements

and representing the configuration of items. Engineering roles are based upon approval authority of technical baseline information and include the following: Design Authority, System Engineer, Project Engineer, Process Engineer, Nuclear Safety and Licensing Engineer, and Design Agent.

2.2.2.1.3 Environment, Safety, Health and Quality

Environment, Safety, Health and Quality includes Environmental, Safety and Health, Quality Assurance, and Radiological Control. The *Integrated Environment, Safety and Health Management System Plan* (RPP-MP-003) establishes a single, defined safety and environmental management system that integrates environment, safety, and health (ES&H) requirements into the work planning and execution processes to effectively protect the workers, public, and the environment. All project participants share in the responsibility to effectively implement RPP-MP-003; however, the ES&H group is to ensure the plan is effectively implemented.

The ES&H group is also responsible for all permitting tasks required to support project design/installation and system operation, including: preparing Notice of Construction applications and supporting documentation to meet *Clean Air Act* and "Washington Clean Air Act of 1967" permit requirements; submittal of modifications to the Hanford Site single-shell and double-shell tank RCRA Part A and Part B permit applications; and preparing pre-operational monitoring determination documentation for affected facilities before installation.

Quality Assurance responsibilities involve verification that all project tasks are performed in accordance with applicable quality assurance requirements. Activities will include procedure reviews and acceptance, audits, surveillances, assessments, design reviews, design verification activities, corrective action reporting, and trend analysis.

Radiological Control is responsible for ensuring that radiological controls are effectively incorporated into the design of the TRU Waste Project. Radiological Control is also responsible for supporting activities performed within the tank farms in preparation for and during tank waste retrieval and packaging operations.

2.2.2.1.4 Project Controls

Project Controls is responsible for tracking the scope, schedule, and budget for all TRU Waste Project elements. Schedule and cost variances are reported to management for corrective action as needed. Additionally, Project Controls is responsible for supporting the procurement of labor, services, subcontractors, material, and equipment.

2.2.2.1.5 Operations and Maintenance

As construction and installation of each tank retrieval system and WPS is completed, the systems will be accepted by operations for use. Plant operations and maintenance will be responsible for completing the retrieval of each tank and packaging the waste for disposal at WIPP.

Operations and Maintenance staff will participate in design reviews, value engineering sessions, technical baseline development, operating and maintenance procedure

development, operational test review and approval, operations acceptance checklist review and approval, training, etc., to ensure new equipment is effectively designed, fabricated, installed, and tested.

2.3 ACQUISITION STRATEGY

The acquisition strategy for TRU Waste Project includes the approach to procure and contract administration approach for system test and evaluation as described in the following subsections.

2.3.1 System Acquisition Approach

The acquisition strategy for TRU Waste Project is based on procuring products and services via competitive bidding to the greatest extent possible, while ensuring that the aggressive TRU Waste Project schedule is maintained. The detailed acquisition strategy will be detailed and implemented at the subproject level as specified in project execution plans.

2.3.2 Startup and Testing

Testing of the TRU Waste Project equipment will be performed in accordance with test plans submitted by the suppliers and/or developed by the CH2M HILL Startup and Testing Organization, as required per *Start-Up and Testing Program Plan* (TFC-PLN-26). All test plans will be reviewed and approved by the CH2M HILL Startup and Testing Organization before start of the test activities. All appropriate test activities will be witnessed by CH2M HILL Startup and Testing representatives and/or quality assurance/third-party inspection personnel.

2.3.3 Readiness Review Process

The TRU Waste Project will undergo a Readiness Review in accordance with *Startup and Restart of Nuclear Facilities* (DOE O 425.1B), and *Readiness Review Program Plan* (TFC-PLN-16).

3.0 RESOURCE REQUIREMENTS

The TRU Waste Project will be performed to established and controlled schedule and cost baselines that are tracked according the project's work breakdown structure (WBS).

3.1 WORK BREAKDOWN STRUCTURE

The TRU Waste Project WBS is distributed within the company WBS. The WBS applicable to the retrieval, packaging, and CSS projects is provided in Table 3. Additional project-specific WBS elements are at the discretion of the respective project managers.

Table 3. Transuranic Waste Project Work Breakdown Structure

Project	WBS	WBS Title
Retrieval	5.8.6.3.13	B-201 Retrieval/Closure
	5.8.6.3.14	B-202 Retrieval/Closure
	5.8.6.3.15	B-203 Retrieval/Closure
	5.8.6.3.16	B-204 Retrieval/Closure
	5.8.3.7.10	T-110 Retrieval/Closure
	5.8.3.7.11	T-111 Retrieval/Closure
	5.8.3.7.13	T-201 Retrieval/Closure
	5.8.3.7.14	T-202 Retrieval/Closure
	5.8.3.7.15	T-203 Retrieval/Closure
	5.8.3.7.16	T-204 Retrieval/Closure
Packaging	5.9.2.2.1	CH TRU Packaging
Characterization, Storage, and Shipping	5.9.2.2.1	CH TRU Packaging

CH = contact-handled.
TRU = transuranic.
WBS = work breakdown structure.

3.2 SCHEDULE BASELINE

The project schedule continues to develop as the project matures. The schedule will be maintained in the integrated mission execution schedule, and as the project matures the schedule will be modified accordingly. The schedule will support the schedule objectives (Section 1.3).

The integrated mission execution schedule baseline schedule will depict project activities and milestones based on project requirements, including regulatory compliance, programmatic, operational, financial, and institutional constraints at a level of detail that enables the project managers and the strategic planning director to evaluate trends monthly and take necessary corrective action. Agreement on the schedule will be reached with the three project managers and the strategic planning director.

3.3 COST BASELINE

The TRU Waste Project is expense-funded. Monthly reports are provided to ORP, including cost performance information. Formal reviews of expense funding are performed by ORP at mid-year. Forecasted expense funding under runs are returned to the program for other uses. Increases in annual expense funding are obtained via change request.

The TRU Waste Project is maintained in the integrated mission execution schedule. Table 4 identifies the funding baseline for the TRU Waste Project.

Table 4. Fiscal Year Funding Baseline for the Contact-Handled Transuranic Waste Project

Fiscal Year	2003	2004	2005	2006	2007
Funding ($M)	5	20	25	8	3
Total ($M)					61

No contingency shall be established for expense-funded projects. Should new requirements be identified, or unanticipated events occur which result in significant cost variances, discussions will be held with ORP to reach agreement on what can be eliminated, deferred or modified to address the issues. Any necessary contract change requests will be expedited.

4.0 PROJECT MANAGEMENT

The TRU Waste Project has risks and uncertainties associated with its execution. To ensure successful completion of the project, the project will employ various project management tools including:

- Adherence to the established and controlled technical, schedule, and cost baselines
- Risk management through risk identification and targeted mitigation
- Identification and documentation of physical, regulatory, and programmatic interfaces
- Regular collection and reporting of project status
- Adherence to governing project procedures
- Performance in accordance with established quality assurance and safety requirements.

4.1 CHANGE CONTROL

Changes to TRU Waste Project technical, schedule, or cost baselines will be authorized by approval of a baseline change request (BCR) in accordance with *Baseline Change Control* (TFC-PRJ-PC-C-12). Thresholds identified in that procedure will be used by the project. The goals of the baseline change control process include the following:

- Prevent unauthorized or unintended deviations from approved baselines
- Recognize, document, approve, and implement changes
- Control consequences of those changes.

All BCRs will be documented on a BCR form. Changes to project baseline documents will be documented on an Engineering Change Notice per *Engineering Change Control* (TFC-ENG-DESIGN-C-06), and will be made an attachment to the BCR form, if necessary. Each BCR will be assigned a unique identification number.

4.2 RISK MANAGEMENT

An integral part of the project management function is risk management. The TRU Waste Project risk management will be achieved by each of the three project elements (retrieval,

packaging, and CSS). The CSS Project will perform periodic reviews to ensure risk integration across the whole project.

4.3 PROJECT INTERFACES

Defining the TRU Waste Project interfaces is an early responsibility of the Project Manager and is accomplished through ICDs. Interface documents will be established in accordance with HNF-IP-0842, *RPP Administration*, Volume 4, Section 2.8, "Interface Control." The TRU Project physical interfaces are shown in Figure 6. Specific ICDs will be generated to clearly delineate those functions, authorities, and responsibilities that are the responsibility of the following:

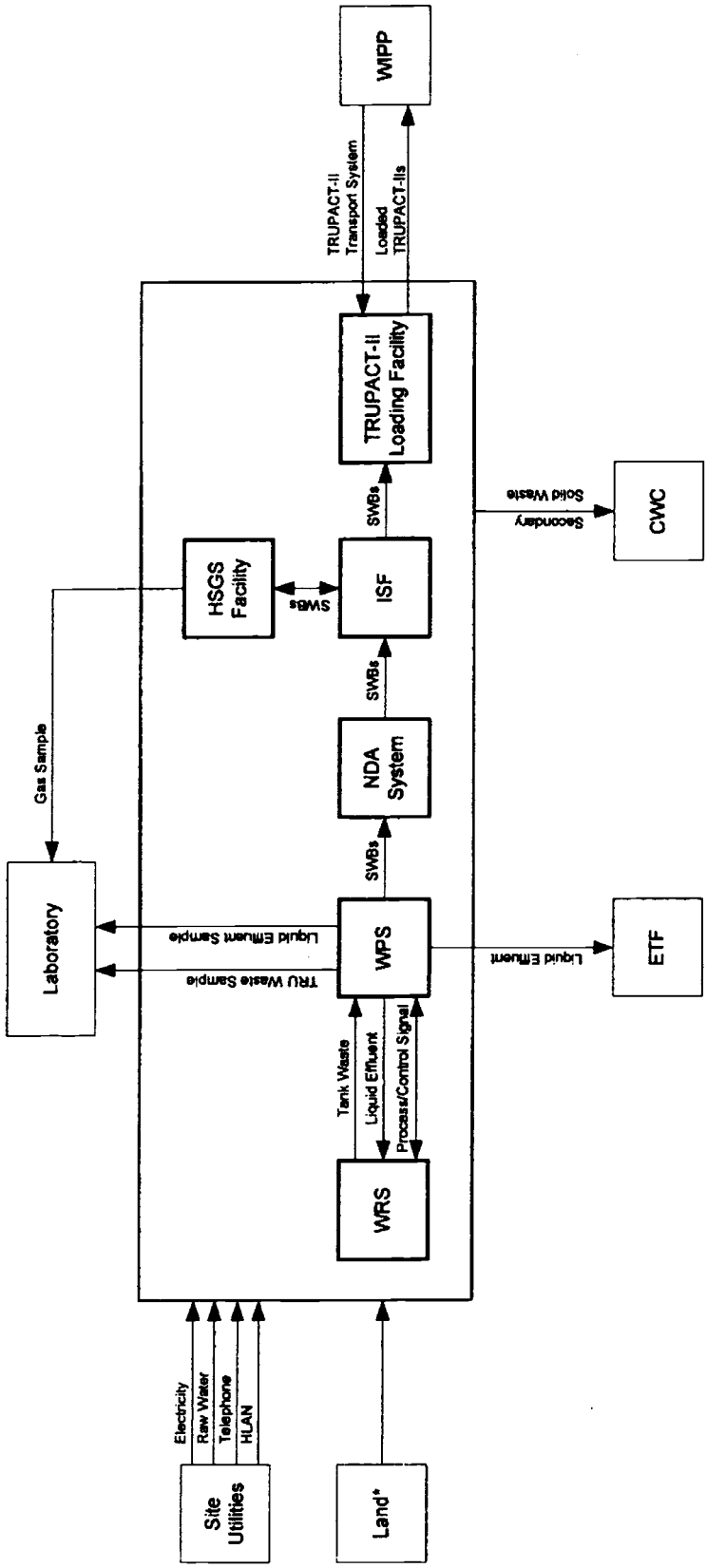
- Retrieval and closure interface (provides feed for packaging operations)
- TRU waste storage interface
- WIPP interface (certification of packaged TRU waste and shipping to WIPP)
- Secondary waste interface (liquid effluent)
- Laboratory interface (sample analysis and sample shipping)
- Site infrastructure (e.g., roads, power, water, telecommunications)
- Material transfers
- TRU Waste Project Manager and the TFC Operations and Maintenance organizations.

These ICDs will be of special importance because they will be basic to establishing the scope of the project. Additional ICDs will be developed as necessary. ICDs are signed by the involved persons, are project baseline documents, and are maintained under change control. The TRU Project life cycle scope is presently planned to extend from feasibility planning through completion of operations and staging of waste for eventual shipment to WIPP. Transition from the project to the operating organization will occur when the structures, systems, and components are declared complete and ready to initiate hot operations.

4.4 PROJECT CONTROLS AND REPORTING

Cost and schedule performance, milestone status, and financial status will be reported to ORP in accordance with *Scheduling* (TFC-PRJ-PC-C-04), TFC-PRJ-PC-C-12, and *Project Control System Description* (RPP-7725) on a monthly basis using ORP-approved WBS elements and data formats. Monthly reporting includes variance analyses. If cost or schedule variances exceed the ORP-established reporting thresholds documented in the Project Control System Description, the report will include corrective action plans that integrate cost, schedule, and scope. Also reported will be analyses of cost and schedule trends, cost and schedule status, baseline change control activity, potential problems, and critical issues.

Figure 6. Transuranic Waste Project Physical Interfaces



*WRS and WPS require siting area within the B and T tank farms.

CWC = Central Waste Complex.
ETF = Effluent Treatment Facility.
HLAN = Hanford Local Area Network.
HSGS = head space gas sampling.
ISF = Interim Storage Facility.
NDA = non-destructive assay.
SWB = standard waste box.
TRU = transuranic.
TRUPACT-II = Transuranic Packaging Transport, Model II.
WIPP = Waste Isolation Pilot Plant.
WPS = waste packaging system.
WRS = waste retrieval system.

Project planning and performance reporting is accomplished by a combination of the Project Manager, Project Controls, and other various performing organizations. The Project Manager has overall responsibility for planning and reporting against the integrated baseline. Supporting subcontractors provide status, via the Project Manager, for work in progress.

Each month the performing organizations report progress to the designated project controls staff, which reviews the reports and forwards them along with earned value data to the project manager. Reports are forwarded to ORP in the form of management review reports. Earned value data are the basis for assessing performance from both a cost and schedule perspective. These data are used in cost control and schedule execution decisions.

4.5 PUBLIC AND STAKEHOLDER INPUT TO PROJECT

Since inception of the TRU Waste Project, an extensive array of public involvement activities has occurred and will continue. Routine briefings to affected stakeholders (e.g., via the Hanford Advisory Board), the tribes, and the New Mexico environmental department have been held. Regulatory activities require public input at various stages (e.g., Notice of Intent public meetings) will occur as a normal part of those functions.

4.6 PROJECT DOCUMENTATION AND RECORDS

Each project is accountable for project documentation and records as specified in their respective project execution plans.

4.6.1 Project Documentation

Procedures and documents supporting the TRU Waste Program must be prepared and maintained in a formalized document control system that complies with the requirements of TFC-PLN-02. The Document Control organization ensures compliance to specific document control requirements by controlling the presentation, modification, review, approval, issuance, distribution and the use of the TRU Waste Program-related documents.

Positive controls will be established for managing the development and maintenance of the WIPP-related documents and procedures. A formalized document control process will be developed by which documents are to be prepared, reviewed, approved, and issued. A separate procedure will be developed to manage the configuration control and change implementation process.

4.6.2 Project Records

TRU Program quality assurance records must be maintained in a formalized and highly structured manner to readily furnish documented evidence that wastes destined for disposal at WIPP are generated, packaged, inspected, assayed, tested, and shipped in a manner compliant with applicable WIPP requirements. Records will be managed to ensure compliance with WIPP-specific record storage and maintenance requirements. The records system will ensure the generation, indexing, classification, retrieval, storage and correction of the TRU Waste Program-related quality assurance records.

Specific procedures will be developed for controlling the generation, sequencing, transfer and storage of WIPP quality assurance records. Records management procedures will stipulate document control and storage requirements for document generators and records management organization as well as formalizing how records are to be prepared, reviewed, approved, and transferred to the records management organization.

4.7 INTEGRATED ENVIRONMENT, SAFETY, AND HEALTH MANAGEMENT SYSTEM

RPP-MP-003 provides the institutional structure for incorporation of ES&H into all aspects of CH2M HILL business and operating units. RPP-MP-003 requirements are implemented through numerous TFC procedures that govern the work planning and execution processes to effectively protect the workers, public, and the environment.

4.8 QUALITY ASSURANCE

The quality assurance requirements that apply to all CH2M HILL activities are identified in *Quality Assurance Program Description for the Tank Farm Contractor* (TFC-PLN-02). TFC-PLN-02 is the source of requirements for CH2M HILL compliance with "Quality Assurance for Nuclear Facilities" (10 CFR 830, Subpart A) and *Quality Assurance* (DOE O 414.1A). TFC-PLN-02 is supported by an implementation matrix, *Quality Assurance Program Description Implementation Matrix* (TFC-ESHQ-Q_ADM-D-04). TFC-ESHQ-Q_ADM-D-04 provides a correlation of regulatory, DOE order, consensus standard, and CH2M HILL quality assurance program requirements, and, it lists documents that serve to implement those requirements. TFC-ESHQ-Q_ADM-D-04 is a tool to be used by project management to identify the processes that implement the various elements of 10 CFR 830, Subpart A, and DOE O 414.1A. The documents listed in TFC-ESHQ-Q_ADM-D-04 (most of which are procedures) determine the quality assurance process for CH2M HILL.

4.9 ENVIRONMENT DOCUMENTATION

The environmental documentation requirements for the TRU Waste Project are as specified in *Environmental Plan to Support Proposed Hanford Tank Waste Supplemental Processing* (RPP-14132). Environmental requirements will be further developed with increased project maturity. Requirements specific to each of the three subprojects will be identified in their respective project execution plans.

4.10 SAFETY AND HEALTH DOCUMENTATION

The current plan is to incorporate the TRU Waste Project into *Tank Farms Documented Safety Analysis (DSA)* (RPP-13033). This process begins with a preliminary hazard categorization that will determine the hazard category of the facility based on projected inventory. Because the facility is receiving tank waste it most likely will be either a Hazard Category 2 or Hazard Category 3 facility and thus must be addressed in RPP-13033. A hazard analysis will then be performed to determine whether the TRU Waste Project poses hazards that are either not already addressed in RPP-13033 or have greater frequencies or consequences than similar hazards identified in RPP-13033. Based on the conclusions of the hazard analysis, any required accident analysis will then be performed. After the accident analysis is complete, any required technical

safety requirement controls will be developed. Finally, a safety basis amendment consisting of RPP-13033 and technical safety requirement (if required) page changes will be generated. This amendment is expected to require ORP approval.

In addition, unreviewed safety question reviews will be performed where required by *Unreviewed Safety Question Process* (TFC-ENG-SB-C-03). When unreviewed safety question review of design media results in a positive unreviewed safety question determination (i.e., not within existing safety basis), either the design will be changed such that it can be implemented within the existing safety basis or the safety basis will be amended.

Industrial and radiological safety will be performed in accordance with approved CH2M HILL procedures.

4.11 CONFIGURATION MANAGEMENT

Configuration management for the TRU Waste Project will preserve and control the technical integrity of products and processes, structures, systems and components, and associated information in accordance with CH2M HILL configuration management standards established in *Configuration Management Plan* (TFC-PLN-23).

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ORIGIN OF WASTES IN SINGLE-SHELL TANKS 241-T-110, 241-T-111 and 241-T-112

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Key Words: Hanford, single-shell tank, 241-T-110, 241-T-111, 241-T-112 Bismuth phosphate process, second plutonium decontamination cycle, 224 building waste, equipment decontamination waste

Abstract: A review of waste transfer documentation was conducted to determine the origin of wastes transferred into single-shell tanks 241-T-110, 241-T-111 and 241-T-112. This review was conducted to support decisions concerning disposition of the waste present in this tank.

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Tank Farm Contractor (TFC) RECORD OF REVISION	(1) Document Number RPP-13873	Page 1

(2) Title
Origin of Wastes in Single-Shell Tanks 241-T-110, 241-T-111 and 241-T-112

Change Control Record

(3) Revision	(4) Description of Change - Replace, Add, and Delete Pages	Authorized for Release (5) Resp. Engr. (print/sign/date)	(6) Resp. Mrg. (print/sign/date)
RS1	Add - Title page: Included tank T-112 in title. Document already discussed tank T-112 waste history. Add - Executive Summary: Included summary of tank T-112 waste history and sum of transuranic elements concentrations. Included discussion on origin of equipment decontamination and coating removal waste. Replace - Executive Summary: Replaced discussion on gross alpha concentrations measurements for tanks waste samples with sum of the transuranic elements with half-lives greater than 20-years. Add - Section 2.2: Expanded discussion on types of records reviewed and information available in each of these records. Add - Sections 2.2.1: Included discussion on prior waste types stored in tanks 241-T-105 and 241-T-106. Add - Section 2.2.3: Discussed variation in sludge volume reported for tank 241-T-110 in historical records. Included discussion on saltwell pumping in 1978 and interim stabilization of tank 241-T-110 in 2000. Add - Section 2.2.4: Indicated that analyses of the equipment decontamination waste solutions were not located during review of available records. Included discussion on saltwell pumping 1976-78 and interim stabilization of tank 241-T-111 in 1995. Add - Section 2.2.5: New section added to expand discussion on cesium ion exchange process waste that was added to tank T-112 in 1973. Included discussion on prior waste types stored in tanks 241-T-105 and 241-T-106 Add - Section 2.2.6: Discussed that WHC-MR-0132 does support waste history described in RPP-13873 for tanks T-111	Michael E Johnson 11/4/04 Michael E. Johnson	SMackay 12/15/04. STEWART MACKAY

Tank Farm Contractor (TFC) **RECORD OF REVISION (continued)**	(1) Document Number RPP-13873	Page 2

Change Control Record

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		(5) Resp. Engr. (print/sign/date)	(6) Resp. Mrg (print/sign/date)
	and T-112. Add - Section 3.1: Added discussion that plutonium precipitate separated from uranium and fission products was washed three times and the wash water combined with the uranium and fission product solution. Add - Section 3.1: Spent nuclear fuel reprocessing completed in the 221 BiPO4 process when plutonium was separated from the metal waste. Add - Section 3.1: Discussed that after October 20, 1954, the CW from 221-T Plant was transferred as a separate waste stream to underground storage tanks. The 1C waste was treated in 221-T Plant to separate fission products using ferrocyanide precipitation and sent to tanks separate from the CW stream. Change - Table 3: Sr concentrations reported for samples from tanks T-108 (top), T-108 (bottom), C-112 and average for 1C/CW were incorrectly transcribed from the reference document. Error was corrected. Add - Section 3.1.1: Included discussion on off-gas scrubbers and silver chemical reactors that were installed in the 221 BiPO4 Plants. Add - Section 3.3: Included new section discussing processing activities conducted in the 221-B Plant from 1961 through 1985 that generated cesium ion exchange process waste solution, as well as other wastes. Add - Section 4.0: Indicated analyses of core samples and waste templates are the basis for the gross alpha and transuranic elements concentrations reported. Add - Section 5.0: Included summary of tank T-112 waste history and sum of transuranic elements concentrations.		

RPP-13873
Revision 1

ORIGIN OF WASTES IN SINGLE-SHELL TANKS 241-T-110, 241-T-111 AND 241-T-112

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EXECUTIVE SUMMARY

A review of waste transfer documentation was conducted to determine the origin of waste transferred into single-shell tanks 241-T-110, 241-T-111 and 241-T-112. This review was conducted to support decisions concerning disposition of the waste present in these tanks.

Tank 241-T-110 received second decontamination cycle (2C) waste from processing plutonium solutions at the 221-T Bismuth Phosphate plant from January 1945 through December 1954, 221-T Plant low activity cell drainage waste from June 1951 through December 1954, and 224-T Concentration building wastes from May 1952 through December 1954. Tanks 241-T-111 and 241-T-112 received 2C waste from the 221-T Plant from January 1945 through October 1956, 221-T Plant low activity cell drainage waste from June 1951 through October 1956, 224-T Concentration building wastes from May 1952 through October 1956, and 221-T Plant equipment decontamination waste from December 1959 through June 1967. Tank 241-T-112 continued to receive 221-T Plant equipment decontamination waste until June 1973. Tank 241-T-112 also received a mixture of coating removal waste 221-B Plant cesium ion exchange process waste from tank 241-T-106 in March 1973.

The second decontamination cycle and 224-T building wastes originated from purification of plutonium solutions. The second decontamination cycle and 224-T building wastes are not waste originating from separating fission products from the uranium fraction of irradiated reactor fuel. Equipment decontamination wastes originated from removing residual radionuclides from failed process equipment to enable this equipment to be repaired and returned to service. Coating removal waste originated from dissolution of the aluminum coating present on irradiated fuel elements, prior to the dissolution of the fuel elements. Cesium ion exchange process waste originated from processing waste solutions at the 221-B Plant to separate cesium from these wastes.

The concentrations of the transuranic elements with half-lives greater than 20-years (i.e. sum of neptunium-237, plutonium-238, plutonium-240, plutonium-240 and americium-241) in the waste stored in tanks 241-T-110, 241-T-111 and 241-T-112 (sludge fraction only) are approximately 83.3ηCi/g, 186.5ηCi/g and 255.2ηCi/g, as reported on October 11, 2004 from the TWINS database.

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LIST OF TERMS

1C	first cycle of the decontamination process
2C	second decontamination cycle
CW	Coating waste
DOE	U.S. Department of Energy
lbs	pounds
MW	Metal waste
REDOX	Reduction-Oxidation
ηCi/g	nanocuries per gram
μCi/cc	microcuries per cubic centimeters
μCi/g	microcuries per gram
μg/cc	micrograms per cubic centimeters

1.0 INTRODUCTION

The origin of the wastes in tanks 241-T-110, 241-T-111 and 241-T-112 is important in determining the disposition of these wastes and the waste storage tanks. Section 2.0 discusses the origin of waste transferred into and removed from single-shell tanks 241-T-110, 241-T-111 and 241-T-112. Section 3.0 provides a description of the different types of wastes that were generated at the Hanford Site chemical processing plants and transferred to these underground storage tanks. Section 4.0 provides a discussion on the transuranic radionuclide analyses of the wastes in these tanks. The concentration of transuranic radionuclides present in these wastes is important to determining the disposition of these wastes. Section 5 summarizes the waste types that were transferred into tanks 241-T-110, 241-T-111 and 241-T-112.

2.0 WASTE TRANSFER INTO AND WASTE REMOVAL FROM TANKS 241-T-110, 241-T-111 AND 241-T-112

This section provides a brief description of tanks 241-T-110, 241-T-111 and 241-T-112 and summarizes waste transfers into and waste removal from these tanks. In order to determine the origins of the wastes presently stored in tanks 241-T-110, 241-T-111 and 241-T-112, publicly available reports for the Hanford Site were reviewed.

Documents reviewed included the Hanford site contractors' monthly reports (1945 through 1975), Army Corp of Engineers monthly reports (December 1944 through December 1946), U. S. Atomic Energy Commission monthly reports (1947 through 1954), waste disposal reports (1948 through 1975), tank farm waste status summary reports, and miscellaneous letters and technical reports.

The Hanford site contractors' monthly reports for January 1945 through July 1951 list the volume of waste stored in the single-shell tanks, with the exception of the B-200 and T-200 series single-shell tanks. No records were located that provided the volume of wastes stored in the single-shell tanks from August 1951 through February 1952. Beginning in March 1952, waste transfers and the volume of waste stored in each single-shell tank were reported for each tank in a waste status summary report.

With the exception of the waste status summary reports, all reports cited in this section are available electronically from the Hanford Declassified Document Retrieval System at http://www2.hanford.gov/declass/ or the U.S. Department of Energy (DOE) Information Bridge at http://www.osti.gov/bridge/. The waste status summary reports are available only as photocopies from Hanford Site Records Information Management Services organization.

2.1 DESCRIPTION OF TANKS 241-T-110, 241-T-111 AND 241-T-112

Single-shell tanks 241-T-110, 241-T-111 and 241-T-112 were originally constructed in 1944 as part of the Manhattan Project (HW-10475-C, chapter IX) and are three of the twelve, 100-series tanks in 241-T Tank Farm. The 100-series tanks are seventy-five-foot diameter underground

tanks made of reinforced concrete with a steel liner on the bottom and sides. Each tank has a design capacity of 530,000 gallons at a liquid depth of sixteen-feet. The overflow pipe for tanks 241-T-110 and 241-T-111 is at an elevation that results in seventeen-feet of waste (~540,530 gallons) being retained in each tank. The overflow pipeline from tank 241-T-112 is at an elevation that results in eighteen-feet of waste (~573,530 gallons) being retained in this tank (HW-27035).

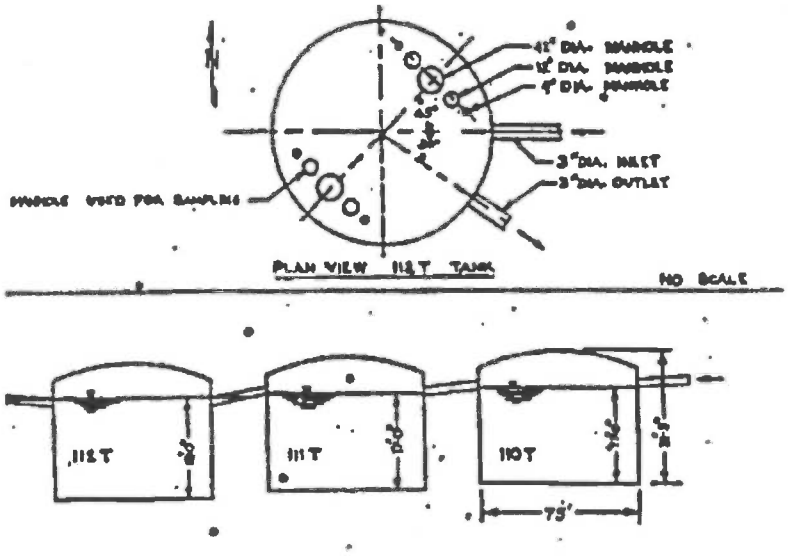
Tanks 241-T-110 and 241-T-111, along with tank 241-T-112, were connected together via underground piping to allow solution to cascade from the lead tank into the subsequent two tanks. Solids settled in each tank, with the supernatant discharged from tank 241-T-112 through an underground pipeline to a crib. In addition to the overflow piping, each tank is equipped with four, 3-inch diameter stainless steel inlet pipes. Originally, only the inlet pipes from tank 241-T-110 were connected to diversion box 241-T-153, with the inlet pipes for the other tanks blanked off close to each tank (HW-10475-C, page 907 –908).

2.2 WASTE TRANSFERS

This section describes waste transfers into and waste removal from tanks 241-T-110, 241-T-111 and 241-T-112. These tanks were operated for a number of years as a three-tank cascade. This section includes a discussion of waste discharge to underground cribs. The design of the tank cascade system is shown in Figure 1 and resulted in tanks 241-T-110 and 241-T-111 being filled with waste that then cascaded into tank 241-T-112. Figure 1 does not represent the current configuration of piping for these tanks. From 1947 through 1951, a jet was used to transfer waste from tank 241-T-112 to the crib. After modifying the disposal system in May 1951, waste was allowed to gravity overflow from tank 241-T-112 to the crib.

The volume and radioactive (plutonium, gross beta, and uranium) content of waste discharged from these tanks to underground cribs is summarized in references HW-17088, HW-20583, HW-25301, HW-28121, HW-33591, HW-38562, HW-44784, HW-72956, ISO-98, and ARH-1608. Appendix A provides a tabular listing of the volume of solids and total waste present in tanks 241-T-110, 241-T-111, and 241-T-112 for January 1945 through December 1975, after which these tanks were no longer used to receive wastes.

Figure 1. Tanks 241-T-110, 241T-111, 241-T-112 Waste Tank Cascade System



2.2.1 Second Decontamination Cycle (2C) Waste

The 241-T Tank Farm was originally constructed to receive waste from the 221-T Bismuth Phosphate plant (see Section 3.0). Tanks 241-T-110, 241-T-111, and 241-T-112 were operated as a cascade. Chemical tracer runs (non-radioactive) were initiated in the 221-T Plant in December 19444, with the second decontamination cycle (designated as 2C) waste from these runs received into tank 241-T-110 (HAN-45800-DEL, page 1). According to the Army Corps of Engineers report for January 1945 (HAN-45800-DEL, page 4), the first radioactive waste was received into tank 241-T-110 from the processing of six charges of material from 100-B reactor in the 221-T Plant to separate plutonium.

Tanks 241-T-110, 241-T-111, and 241-T-112 continued to receive 2C waste through July 22, 1946, at which time these tanks were reported as being filled and 2C waste was diverted to tanks 241-T-105 and 241-T-106 (HAN-45800-DEL, page 64 and HW-7-4542-DEL, page 21). Tanks 241-T-105 and 241-T-106 were originally designated as a spare set of tanks for receipt of 2C waste from the 221-T Plant. In order to allow the collection of 2C waste in tanks 241-T-105 and 241-T-106, a separate transfer pipeline was established to the inlet of tank 241-T-105 on July 17, 1946 (H-2-578 and HAN-45762, pages 27 and 32).

While tanks 241-T-110, 241-T-111, and 241-T-112 remained filled with 2C waste, tanks 241-T-105 and 241-T-106 continued to receive 2C waste from the 221-T Plant. Measurements of the solids depth in tanks 241-T-110, 241-T-111, and 241-T-112 were conducted in October 1946 using an ionization chamber indicated that only tank 241-T-110 contain solids, evenly distributed at a depth of approximately 38 inches, corresponding to ~84,030 gallons (H-7-5362-DEL, page 27).

Plans were initiated in October 1946 to dispose of the 2C supernatant contained in these tanks to an underground crib (HW-7-5362-DEL, page 27). A new underground crib (designated as 241-T-3) was constructed in 1947. Tank 241-T-110 would be used to settle solids that formed in the 2C waste, with the supernatant cascading by gravity flow into tank 241-T-111 and then into tank 241-T-112. The clarified 2C supernatant would be jetted from tank 241-T-112 to the underground crib. Crib disposal of the clarified 2C supernatant was authorized on an experimental basis (HW-10321). The 2C waste contained in tank 241-T-111 was jetted to this underground crib in September 1947 (HW-7795-DEL, page 26).

As part of the planned disposal of the 2C supernatant to the underground crib, separate waste transfer lines were routed to tanks 241-T-111 and 241-T-112 (see drawing H-2-578). This would enable filling these tanks directly with 2C waste when tank 241-T-110 filled with solids and was no longer suitable as a settling tank. Approximately 20,000-gallons of 2C supernatant were jetted from tank 241-T-112 to the underground crib in November 1947 to enable a waste transfer line tie-in from diversion box 241-T-153 to tanks 241-T-111 and 241-T-112 (HW-8267-DEL, page 27). Crib disposal of additional 2C supernatant was delayed until a means to sample the soil in dry wells that surround the crib area was developed.

A tool for sampling the soil in dry wells surrounding the 2C disposal crib area was designed, constructed, and tested in February 1948 (HW-9191-DEL, page 28), but this tool proved unsuccessful in obtaining soil samples when used in March 1948 (HW-9595-DEL, page 30). However, approval was given to resume limited crib disposal of 2C supernatant in April 1948, since tanks 241-T-105 and 241-T-106 were nearly filled with 2C waste and additional storage space in the single-shell tanks was not available. Crib disposal of approximately 360,000 gallons of 2C waste from tank 241-T-105 was conducted in April 1948 (HW-9922-DEL, page 31).

Following extensive sampling of the soil surrounding the 2C waste disposal crib (HW-10166-DEL, page 31), crib disposal of 2C waste contained in tank 241-T-106 was conducted in July 1948 (HW-10714-DEL, page 32) and August 1948 (HW-10993-DEL, page 32). Crib disposal of approximately 450,000 gallons of the 2C waste in tank 241-T-112 was initiated on August 4, 1948 (HW-10993-DEL, page 35) and stopped in September 1948 (HW-11226-DEL, page 32) to allow installation of an experimental sand filter on the jet discharge from tank 241-T-112 to the crib. The experimental sand filter was installed to determine the feasibility of removing additional activity from the 2C supernatant being disposed to the crib. Crib disposal of the remainder of the 2C waste in tank 241-T-112 was completed in October 1948 (HW-11499, page 33).

With the emptying of tank 241-T-112 in August through October 1948, 2C waste was again routed from the 221-T Plant into the cascade of tanks 241-T-110, 241-T-111, and 241-T-112 beginning in August 1948. Tank 241-T-110 was used to settle solids that formed in the 2C waste, with the supernatant cascading by gravity flow into tank 241-T-111 and then into tank 241-T-112. The clarified 2C supernatant was periodically jetted from tank 241-T-112 to the crib (HW-33591, pages 4 and 26) from August 1948 through May 1951. In May 1951, modifications were conducted that allowed the 2C supernatant waste to gravity overflow from tank 241-T-112 into the crib (HW-21260-DEL, page 57).

2.2.2 2C Waste Combined with Cell Drainage Waste

Beginning in June 1951, the neutralized, cell drainage waste from the 221-T Plant (designated as 5-6 waste) was combined with the 2C waste in the cascade of tanks 241-T-110, 241-T-111, and 241-T-112 (HW-21506-DEL, page 56 and H-2-1988). Tank 5-6 in the 221-T Plant was used to collect low activity drainage from the process cells. The generation of cell drainage waste was intermittent and dependent on the frequency of leaks that developed in the 221-T Plant process equipment. High-activity cell drainage waste was collected in tank 5-9 and either reworked or transferred to single-shell tank 241-T-107 (see Section 3.1.1).

The low activity cell drainage was transferred to the cascade of tanks 241-T-110, 241-T-111, and 241-T-112 so "... that the major portion of the suspended plutonium carrying solids will settle out while the waste solution combines and cascades concurrently with the second decontamination cycle waste prior to underground cribbing by constant overflow" (HW-21506-DEL, page 56). The combined 2C waste and cell drainage waste from tank 5-6 were transferred to the cascade of tanks 241-T-110, 241-T-111, and 241-T-112. All three tanks were essentially filled with waste to the overflow pipeline. Solids gravity settled and supernatant gravity overflow from tank 241-T-112 into the crib.

2.2.3 2C, Cell Drainage, and 224-T Concentration Building Waste

Beginning on May 29, 1952, the waste from the 224-T Concentration building (designated as 224 waste) was discharged to the cascade of tanks 241-T-110, 241-T-111, and 241-T-112 along with the cell drainage waste collected in tank 5-6 and the 2C waste from the 221-T Plant (HW-27838, page 17). Section 3.1 provides a description of the plutonium concentration process conducted in the 224-T Concentration building. These three waste streams (2C / 224 / 5-6) continued to be collected in the cascade of tanks 241-T-110, 241-T-111, and 241-T-112. All three tanks were essentially filled with waste to the overflow pipeline. Solids gravity settled and supernatant gravity overflow from tank 241-T-112 into the crib.

In December 1954, tank 241-T-110 was reported as filled with sludge (530,000 gallons) and only tanks 241-T-111 and 241-T-112 were receiving the 2C / 224 / 5-6 waste streams (HW-34412, page 6 and H-2-2398). A review of Hanford Site monthly reports and waste status summary reports from 1955 to the present indicate that no additional waste was transferred into tank 241-T-110. This review also observed that the documented, volume of solids contained in this tank was recorded generally as 530,000 gallons through July 1957and then reported as 46,000 gallons from August 1957 through June 1966. The tank 241-T-110 sludge volume was reported typically as 508,000 gallons from July 1966 through September 1969, then 293,000 gallons through September 1974, and 466,000 gallons from October 1974 through March 1982. Following completion of salt well pumping in 1978 (RMIS TFIC # D196235596), the tank 241-T-110 sludge level was reported at 370,000 gallons in April 1982 to the present. Supernatant and interstitial liquids were removed from tank 241-T-110 in 2000 as part of interim stabilization of the single-shell tanks (HNF-SD-RE-TI-178, pages 218-222).

The reason for the variations in the measured solids volume in tank 241-T-110 was not reported in the waste status summary reports, but could be due to inaccurate measurements. Sludge depth measurements were obtained by lowering a weight through a riser in the single-shell tank and attempting to determine the sludge interface with the supernatant. The distance from the riser bench-mark to the sludge interface was determined and the sludge depth was then calculated. The date when sludge measurements were actually obtained was not located in the available documentation. An erroneous measurement or calculation may have been recorded in August 1957 and the sludge depth not measured again until July 1967. Sludge removal from tank 241-T-110 has not been conducted as of June 2004 and is not the source for the variation in measured sludge volume.

Tanks 241-T-111 and 241-T-112 continued receiving the 2C / 224 / 5-6 waste streams and by March 1955, were reported as containing approximately 487,000-gallons and 33,000 gallons of sludge, respectively (HW-36001, page 6). This prompted the transfer in April 1955 of 115,000 to 133,000-gallons of sludge from tank 241-T-111 to tank 241-T-112 (HW-36553, page 6). Tank 241-T-111 was reported as containing approximately 362,000 gallons of solids after this transfer. Tank 241-T-112 was reported as having 33,000 gallons of solids before this transfer (HW-36001, page 6) and approximately 170,000 gallons of solids after this transfer (HW-37143, page 6). The solids were transferred from tank 241-T-111 into tank 241-T-112 to provide sufficient space in tank 241-T-111 for gravity settling of solids present in the 2C / 224 / 5-6

wastes before the clarified supernatant was overflowed to tank 241-T-112 and to the 241-T-3 crib (after 1958 referred to as the 216-T-7 crib).

The 241-T-3 crib continued to receive the supernatant overflowed from tank 241-T-112 until November 30, 1955, after which the 241-TX-153 crib (after 1958 referred to as the 216-T-19 crib) was used (HW-44784, pages 43 and 44). Additionally, approximately 700,000 gallons of waste was discharged from tank 241-T-112 to the 241-T trench number 5 on May 5, 1955, to empty this tank (HW-38562, page 28). Trench 241-T number 5 is also referred to as trench number 216-T-5 (HW-48518, page 42).

The 2C / 224 / 5-6 wastes continued to be transferred into the cascade of tanks 241-T-111 and 241-T-112 through March 20, 1956, when the final processing of irradiated fuels for plutonium recovery was completed in the 221-T Plant (HW-42219-DEL, page ED-5). Process equipment flushes using nitric acid and peroxide – caustic were conducted in the 221-T Plant from March 1956 (HW-42219-DEL, page ED-5) through September 1956 (HW-45707-DEL, page D-5) to recover plutonium and remove fission products from the equipment. The acid flushes were processed through the normal flowsheet, generating additional 2C and 224 wastes that were transferred to the cascade of tanks 241-T-111 and 241-T-112. The 221-T Plant was placed in standby status whereas the 224-T building was placed in lay-away status in October 1956 (HW-46432-DEL, page D-5). The volume of solids and liquid report in tanks 241-T-111 were 510,000 gallons and 20,000 gallons as of September 30, 1956 (HW-45738, page 6). The volume of solids and liquid report in tanks 241-T-112 were 170,000 gallons and 259,000 gallons as of September 30, 1956 (HW-45738, page 6).

Water transfers through the equipment in the 221-T Plant were conducted once per week beginning in October 1956 following chemical flushing to keep the gaskets installed in piping wetted (HW-46432-DEL, page D-5). If the gaskets dried out, leaks could develop if the equipment were restarted. Water transfers through the 221-T Plant equipment were continued through January 1957 (HW-48132-DEL, page D-6) and were terminated when the 221-T Plant was transitioned to final lay-away status in June 1957 (HW-51211-DEL, page D-6).

The disposition of the water transferred through process equipment in the 221-T Plant is not specified in the Hanford Site monthly reports or waste status summary reports. Reports that document radioactive liquid discharges to the ground for 1956 through 1959 do not indicate the discharge of any waste from tank 241-T-112 to the crib (number 241-TX-153 also known as the 216-T-19 crib) after August 1956 (HW-48518, page 35, HW-59359, page 7, and HW-63646, page 7). Tank 241-T-111 was filled to the overflow pipeline and the total waste volume in tank 241-T-112 fluctuated from 429,000 gallons (HW-45738, page 6) to 417,000 gallons (HW-50127, page 6) during this period, without any cause noted for the volume changes. Therefore, it cannot be determine with certainty whether the water used to wet equipment in 221-T Plant was discharged to tank 241-T-111 and 241-T-112.

2.2.4 Equipment Decontamination Waste

The 221-U Plant was being used to decontaminate equipment from the Reduction-Oxidation (REDOX) plant, which processed spent nuclear fuels to recover uranium and plutonium. In October 1958, plans were developed to convert the 221-T Plant for use as decontamination facility for equipment from the REDOX plant (HW-58051-DEL, page D-5) and use the 221-U Plant for another purpose. Work was conducted from February 1959 (HW-59434-DEL, page D-4) through June 1960 (HW-65935-DEL, page C-2) to convert the 221-T Plant to an equipment decontamination facility. Equipment decontamination activities were initiated at the 221-T Plant in July 1960, with the receipt of a failed multipurpose dissolver from the REDOX plant (HW-66271-DEL, page C-2).

The Hanford Site monthly reports and waste status summary reports indicate that no waste was transferred into or out of tanks 241-T-110, 241-T-111 and 241-T-112 from August 1956 through November 1959 during modifications to the 221-T Plant. In December 1959, 2,750 gallons of waste were transferred from 221-T Plant into tank 241-T-111 (HW-83906-C-RD, page 92), presumably resulting from the equipment modifications conducted at 221-T Plant. The composition or specific source of the equipment modification waste was not found during review of available documentation. However, all 221-T Plant equipment had been flushed using nitric acid and peroxide – caustic solution during 1956 to recover plutonium and remove fission products (see section 2.2.3). Therefore, the waste transferred from 221-T Plant to tanks 241-T-111 and 241-T-112 would have contained only residual levels of fission products.

As part of readying 221-T Plant for this new mission, a route was established in November 1959 from the 221-T Plant to crib number 241-TY (later referred to as 216-TY-3 or 216-T-28) for disposal of low activity waste (HW-62864, page D-4). Low activity waste was transferred from 221-T Plant into the cascade of tanks 241-T-111 and 241-T-112 and then pumped from tank 241-T-112 to the underground crib. The waste status summary reports for the underground storage tanks at the Hanford Site indicate tank 241-T-112 received 3,000 gallons of waste from 221-T Plant in March 1960 (HW-64810, page 6 and HW-83906-C-RD, page 119) and 16,000 gallons in May 1960, with 44,000 gallons of waste pumped to the 241-TY-3 crib (HW-65643, page 6 and HW-83906-D-RD, page 131). Additional decontamination waste continued to be received periodically into the cascade of tanks 241-T-111 and 241-T-112 and was pumped to the underground crib (216-T-28; then 216-T-36 after May 1967) through June 1967 (HW-83906-D-RD, HW-83906-E-RD, ISO-538, and ISO-674).

After July 1967, equipment decontamination waste from 221-T Plant was transferred directly into tank 241-T-112, with the supernatant discharged to crib number 216-T-36 (ARH-95). Tank 241-T-111 no longer was used to receive waste. Supernatant and interstitial liquids were removed from tank 241-T-111 between 1976 and 1978 (RMIS TFIC #D196235379) and 1995 as part of isolation and interim stabilization of the single-shell tanks (HNF-SD-RE-TI-178, pages 223-225).

From July 1967 through June 1972, equipment decontamination waste was transferred from 221-T Plant into directly into tank 241-T-112. Waste was transferred from tank 241-T-112 to the

REDOX plant for evaporation, with the concentrated waste transferred to other single-shell tanks (ARH-1200 C, ARH-1200 D, ARH-1666 A, B, C, D, ARH-2074 A, B, C, D, and ARH--2456 A, B). From July 1972 through June 1973, equipment decontamination waste was transferred from 221-T Plant into tank 241-T-112, and then to single-shell tank 241-U-107 (ARH-2456 C, D, and ARH-2794 A, B). After June 1973, tank 241-T-112 was no longer used to receive 221-T Plant decontamination waste. The equipment decontamination waste was transferred from 221-T Plant into tank 241-U-107 beginning in October 1973 (ARH-2794 D).

2.2.5 Cesium Ion Exchange Process Waste

Tank 241-T-112 received 350,000 gallons of a mixture of coating removal waste, B-Plant cesium ion exchange waste, and laboratory waste from tank 241-T-106 and 20,000 gallons of waste from diversion box catch tank 241-T-301 in January through March 1973 (ARH-2794A). No other waste was transferred into tank 241-T-112 after March 1973. Supernatant and interstitial liquids were removed from tank 241-T-112 in 1974, 1976 and 1981 as part of isolation and interim stabilization of the single-shell tanks (HNF-SD-RE-TI-178, page 6).

Prior to 1973, tank 241-T-106 was used to stored 2C waste, first decontamination cycle (1C) and coating removal waste (CW) from the 221-T Plant and coating removal waste from the REDOX Plant. As discussed in section 2.2.1, tanks 241-T-105 and 241-T-106 were spare single-shell tanks that were placed in service to receive and stored 2C waste from the 221-T Plant beginning in July 1946. The 2C waste was received into tank 241-T-105 from July 23, 1946 (HW-7-4542-DEL, page 22) through April 1948, after which the 2C supernatant waste was discharged to a crib (HW-9922-DEL, page 31). Tank 241-T-106 began to receive 2C waste through the overflow line from tank 241-T-105 in June 1947 (HW-7-7454-DEL, page 26) and was filled in March 1948 (HW-9595-DEL, page 32). The 2C supernatant waste contained in tank 241-T-106 was discharged to a crib from July 1948 (HW-10714-DEL, page 32) through August 3, 1948 (HW-10993-DEL, page 35).

After emptying the 2C supernatant waste from tanks 241-T-105 and 241-T-106, the combined 1C/CW waste was transferred from 221-T Plant to tank 241-T-105 beginning in May 1948 (HAN-45807-DEL, page 55). Waste began to cascade from tank 241-T-105 into tank 241-T-106 in August 1948. Tank 241-T-105 continued to receive 1C/CW waste and cascade waste into tank 241-T-106 through January 1949, at which tanks 241-T-105 and 241-T-106 were filled with 1C/CW waste (HW-12391-DEL, page 38). The 1C/CW supernatant contained in tank 241-T-106 (along with other tanks in 241-T farm) was processed through the 242-T Evaporator from in 1951 with the concentrated 1C/CW supernatant waste (i.e., evaporator bottoms) stored in tanks 241-TX-116 and 241-TX-117. The evaporator bottoms in tanks 241-TX-116 and 241-TX-117 were eventually processed again through the 242-T Evaporator to further concentrate these wastes for storage in tanks 241-TX-110 and 241-TX-111 (RPP-16129, section 2.2.2). Tank 241-T-106 again was used as part of the cascade of tanks 241-T-104 and 241-T-105 to store 1C/CW waste from the 221-T Plant from August 1951 through September 1954, with some of the 1C/CW supernatant discharged from these tanks to a trench in early 1954 (RPP-16129, sections 2.2.3 and 2.2.4). All of the 1C/CW supernatant was transferred from tank 241-T-106 to 241-TX-

118 in January 1955 for processing in the 242-T Evaporator, leaving approximately 10,000 gallons of 2C and 1C/CW sludge in this tank (HW-35022, page 5).

Tank 241-T-106 then received ~221,000 gallons of REDOX Plant coating removal waste supernatant in May 1956 from tank 241-U-110 (HW-43490, page 5). Tank 241-T-106 next received 221,000 gallons of REDOX Plant coating removal waste supernatant from tank 241-S-107 via the cascade overflow line from tank 241-T-105 in June 1965 (HW-83906-E-RD, page 62c) and an additional 90,000 gallons of this same waste type in 1966 (ISO-226, page 5). The REDOX Plant coating removal waste was transferred from tank 241-T-106 to tank 241-TY-101 in the third quarter of 1969, leaving approximately 26,000 gallons of sludge and 42,000 gallons of supernatant in this tank (ARH-1200 C, page 7).

In January through March 1973, tank 241-T-106 received a mixture of supernatant wastes (~455,000 gallons) from tank 241-T-105 consisting of B-Plant cesium ion exchange waste and laboratory waste (ARH-2794A). Approximately 350,000 gallons of supernatant were then transferred from tank 241-T-106 to tank 241-T-112 in June 1973.

2.2.6 Comparison with Other Reports

Waste transfers into and waste removals from tanks 241-T-110, 241-T-111 and 241-T-112 were summarized in *A History of the 200 Area Tank Farms* (WHC-MR-0132), *Waste Status and Transaction Record Summary for the Northwest Quadrant of the Hanford 200W Area* (WHC-SD-WM-TI-669, Rev. 1), *Historical Tank Waste Content Estimate for the Northwest Quadrant of the Hanford 200 West Area* (HNF-SD-WM-ER-351, Rev. 1), and *Waste Status and Transaction Record Summary (WSTRS) Rev. 4* (LA-UR-97-311). In general, the information cited in Sections 2.2.1 through 2.2.4 is in agreement with these previous reports.

These previous reports accurately state the volume of waste transferred into and removed from tanks 241-T-110, 241-T-111 and 241-T-112, as well as the volume of solids and total waste stored in each tank. Specifically, these previous reports do indicate the waste transferred to this tank cascade from was 2C waste from January 1945 through May 1951, combined 2C / 5-6 waste from June 1951 (WHC-MR-0132, page 4) through 1st quarter 1952 and 2C / 224 / 5-6 waste from 2nd quarter 1952 through 1st quarter 1957. These previous reports do indicate that the source of waste transferred into tank 241-T-111 from December 1959 (4th quarter 1959) through June 1967 was from 221-T Plant, but do not describe these wastes as originating from equipment decontamination conducted in the 221-T Plant (see Section 2.2.4). These previous reports also accurately reflect the waste transfer history associated with tank 241-T-112, as described in this report.

3.0 TYPES OF TANK WASTE GENERATED AT THE HANFORD SITE CHEMICAL PROCESSING PLANTS

There were numerous spent nuclear fuel reprocessing, research and development, plutonium processing and waste management activities conducted at the Hanford Site starting in 1944. These spent nuclear fuel reprocessing, research and development, plutonium processing and waste management activities conducted in the processing plants are discussed further in the DOE/RL-97-02, *National Register of Historic Places Multiple Property Document Form - Historic, Archaeological and Traditional Cultural Properties of the Hanford Site, Washington February 1997.*

It has been established in Section 2.0 that second decontamination cycle (2C) wastes and tank 5-6 cell drainage wastes from the 221-T Bismuth Phosphate plant and 224-T building wastes were transferred into tanks 241-T-110, 241-T-111, and 241-T-112. Additionally, tanks 241-T-111 and 241-T-112 received equipment decontamination waste and tank 241-T-112 received coating removal waste and 221-B Plant cesium ion exchange process waste. The following sections provide a discussion of the wastes originating from operation of the 221-T Bismuth Phosphate plant, 224-T Concentration building and 221-B Plant cesium ion exchange process waste. Equipment decontamination waste from the 221-T Plant was previously discussed in Section 2.2.4.

3.1 B AND T BISMUTH PHOSPHATE PROCESS PLANTS

B- and T-Plants were constructed in 1944 through 1945 to separate plutonium from spent nuclear fuel using the bismuth phosphate process. Figure 2 shows a summary of the 221-B/T Plant bismuth phosphate process, which is referred to throughout this discussion.

In the bismuth phosphate process, the aluminum cladding of spent nuclear fuel elements was dissolved in boiling sodium nitrate solution, to which sodium hydroxide was slowly added (HW-10475-C, page 403). The cladding removal waste, sometimes referred to as coating waste (CW), was transferred to single-shell underground storage tanks (see item [1] in Figure 2).

Reprocessing of the spent nuclear fuel commenced with the dissolution of the uranium fuel elements. The uranium fuel elements (see item [2] in Figure 2) were then dissolved in nitric acid (HW-10475-C, chapter IV, page 405). Water and sulfuric acid were added to the dissolved uranium metal solution and the mixture was then transferred to the plutonium extraction section. The sulfuric acid formed a uranyl sulfate complex that prevented uranium precipitation as a phosphate in the subsequent plutonium extraction step (HW-10475-C, page 418).

Plutonium was extracted from the acid solution by addition of bismuth nitrate and phosphoric acid to form a bismuth phosphate carrier precipitate (HW-10475-C, page 503). The plutonium and bismuth phosphate carrier precipitate was centrifuged and washed three times with water to separate the acidic supernatant from the plutonium precipitate (see item [3] in Figure 2). The acidic solution remaining after the plutonium precipitation contained about 99 percent of the uranium, about 90 percent of the fission products. This separation process also removed and

reduced the gamma radiation activity level in the plutonium precipitate by a factor of 10. However, zirconium is phosphate insoluble and zirconium-95 (10 percent of the activity) stayed with the plutonium product. The acidic uranium solution was then neutralized and transferred to the underground single-shell tanks as metal waste (MW). Recent laboratory testing of the bismuth phosphate flowsheet confirms this partitioning of radionuclides (internal letter 7G300-02-NWK-024, "Bismuth Phosphate Process Radionuclide Partition Factors for the Hanford Defined Waste Model"). The laboratory tests indicate the percentage of cecium-137 and strontium-90 partitioned to the metal waste may have been as high as 100 percent and 89 percent, respectively.

After separating and washing the plutonium precipitate from the metal waste, reprocessing of spent nuclear fuel was completed in the 221 Plant Bismuth Phosphate process. Plutonium decontamination was conducted in the remainder of the 221 Plant Bismuth Phosphate process. The plutonium bearing cake was dissolved in nitric acid and further decontamination of the plutonium to separate fission products was conducted (HW-10475-C, chapter VI). Sodium bismuthate, sodium dichromate, or potassium permanganate was added to oxidize the plutonium to the +6 valence-state. This step caused the bismuth phosphate to precipitate phosphate insoluble fission products ("by-product precipitation"), leaving the plutonium in solution. The precipitate was separated from the plutonium-bearing solution using centrifuges and washed to remove soluble plutonium. The plutonium was reduced to the +4 valence state to form a precipitate that could be separated from the remaining soluble fission products by centrifugation.

The fission products separated from the plutonium product during this first cycle of the decontamination process (designated as 1C) were transferred to single-shell tanks. The 1C waste (see item [4] in Figure 2), contained approximately 10 percent of all fission products and approximately 1.4 percent of the plutonium present in the original fuel charged to the plant (HW-23043, pages 20 and 22). After 1951, the Bismuth Phosphate process flowsheet was modified to include cerium and zirconium scavenger precipitation in the 1C by-product step to remove lanthanide and zirconium radionuclides from the plutonium product (HW-23043, page 16). During operation of 221-B Plant, the 1C waste was combined with the coating removal waste and transferred to the same single-shell tank. This same practice was conducted in 221-T Plant from December 1944 through October 19, 1954. Beginning on October 20, 1954, nickel ferrocyanide scavenging of the 1C waste was conducted in T-Plant (but not in B-Plant) to precipitate cesium-137 and strontium-90 (HW-33585-DEL, page Ed-8, and HW-33184). The precipitated 1C waste slurry was transferred separate from the coating removal waste to different single-shell tanks for settling of the precipitate and discharge of the scavenged (i.e., cesium and strontium depleted) supernatant to a crib.

The plutonium solids were again dissolved in nitric acid. A second decontamination cycle (see item [5] in Figure 2) was conducted to reduced the gamma activity level by a factor of 10,000 from that in the previous dissolved metal solution, giving an overall process decontamination factor of 100,000 below that of the original solution (HW-10475-C, page 627). The second decontamination step essentially repeated the steps previously described for the first cycle decontamination. The second decontamination cycle wastes (designated as 2C) were also transferred to the single-shell tanks. The 2C waste contained less than 0.1 percent of the uranium and fission products and about 0.4 percent of the plutonium present in the original fuel charged

to the plant (HW-23043, pages 26 and 28). The plutonium product from the bismuth phosphate process was subsequently concentrated in the 224-T and 224-B buildings using a lanthanum fluoride precipitation process.

Table 1 provides the flowsheet estimated compositions of the neutralized CW, MW, 1C, and 2C waste solutions generated from the bismuth phosphate plants based on the October 1, 1951 flowsheet (HW-23043). Additional analyses of the supernatant fraction of MW, 1C, and 2C that was stored in single-shell tanks are provided in Tables 2 and 3. The CW was combined with the 1C waste in the same tanks in the Bismuth Phosphate process. Note that the coating waste batch size shown in Table 1 is based on 6,600-lbs uranium, but that the metal waste dissolution batch size is based on 2,200-lbs uranium. These sample analyses support that the 2C waste contained less than 0.1 percent of the fission products. Analyses of the combined 2C / 224 building / tank 5-6 waste supernatant stored in tank 241-T-112 conducted on August 6, 1952 and September 24, 1952 indicate that the total beta emitters was comprised of 35 to 50 percent ruthenium, 35 to 50 percent cesium, 4 to 8 percent cerium, yttrium, and other rare earths, and 6 to 11 percent undetermined (HW-27035, page 8).

Figure 2. Bismuth Phosphate Process Diagram

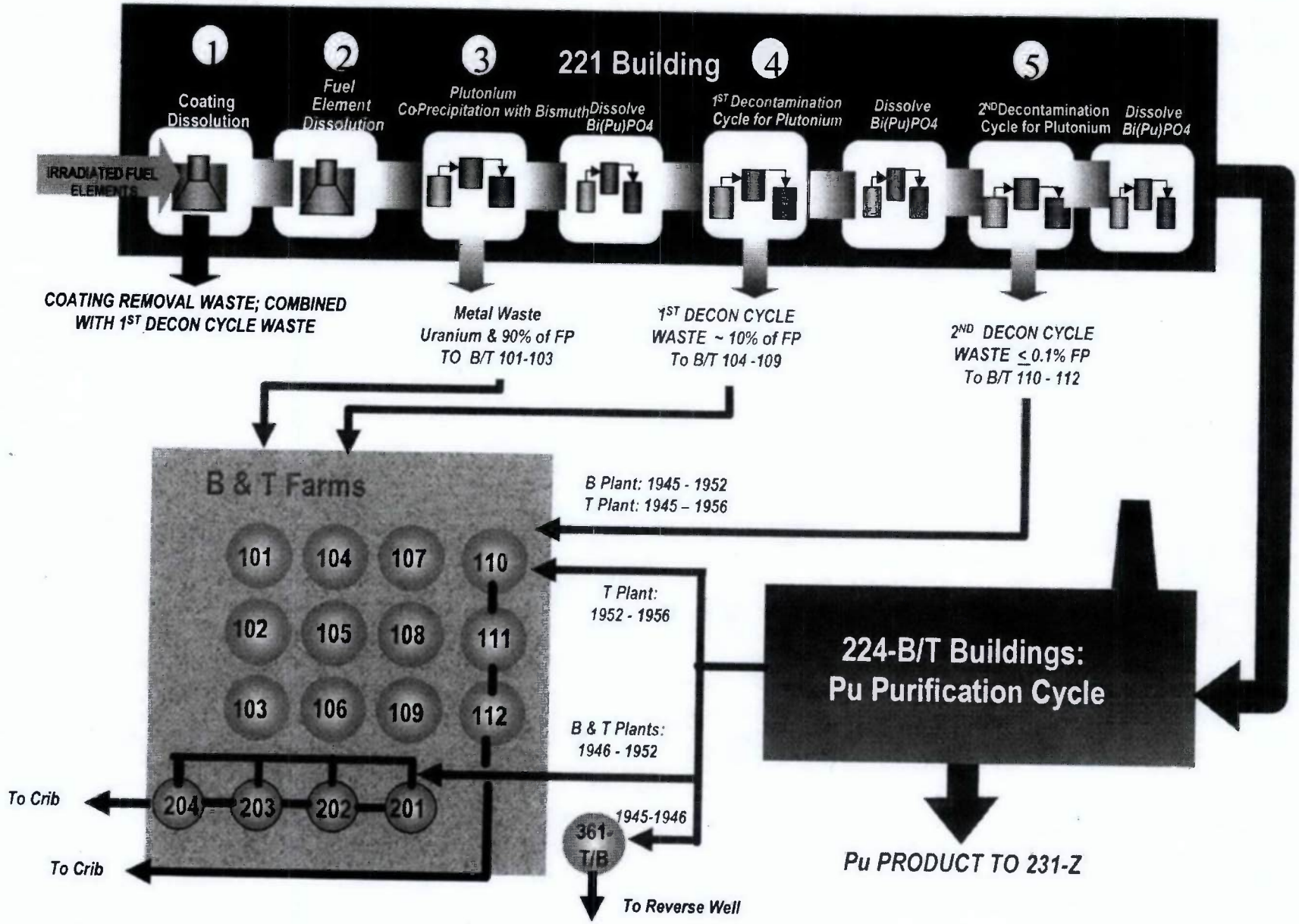


Table 1. Estimated Composition of Bismuth Phosphate Plant Wastes
From October 1, 1951 Flowsheet [1]

Analyte [2]	Coating Removal Waste	Metal Waste	First Decontamination Cycle (1C) Waste	Second Decontamination Cycle (2C) Waste	224 Building Waste
Plutonium	3.3E-04	2.0E-04	6.0E-07 [4]	1.6E-07 [5]	1.68E-04[6]
Uranium	0.15		0.235 [4]	Not reported	2.04E-05
Gamma	6.6E+04	1.3E+07	2.3E+06 [4]	1.13E+04 [5]	1.13E+02[6]
Sodium Aluminate ($NaAlO_2$)	95.1				
Sodium Hydroxide (NaOH)	43.6				
Sodium Nitrate ($NaNO_3$)	61.8				
Sodium Nitrite ($NaNO_2$)	56.0				
Sodium Silicate ($NaSiO_3$)	4.3				
Uranyl nitrate (UHN) [3]		132			
Fluorine (F)					5.6
Nitrate (NO_3)		9.7	93.1	61.3	42.4
Sulfate (SO_4)		24.4	4.73	3.61	0.35
Phosphate (PO_4)		25.2	26.2	23.0	3.05
Sodium (Na)		83.2	47.3	36.7	36.8
Bismuth (Bi)			2.59	1.31	1.18
Cerium (Ce)			0.030		
Lanthanum (La)					0.49
Manganese (Mn)					0.33
Zirconium (Zr)			0.030		
Iron (Fe)			1.37	1.82	
Chrome (Cr)			0.16	0.06	0.17
Ammonia (NH_4)			1.98	1.71	0.12
Silicon Hexa-Fluoride (SiF_6)			4.35	3.67	
Volume per Batch (gallons)	795	2,380	2,040	2,090	2,200

Notes:

[1] See HW-23043

[2] Analyses are reported in grams per liter, except for gamma activity, which is counts/minute/mL.

[3] HW-23043, page 31, notes that uranium is not actually present in this form, but is probably as $NaUO_2PO_4$ and $Na_4(UO_2)_2CO_3$.

[4] Pu and Gamma concentrations were calculated from the compositions of tanks 13-4 and 14-3 (HW-23043, pages 20 and 22).

[5] Pu and Gamma concentrations were calculated from the compositions of tanks 18-4 and 19-3 (HW-23043, pages 26 and 28).

[6] Pu and Gamma concentrations were calculated from the compositions of tanks A-4, D-4, B-3, and F-8 (HW-23043, pages 39, 44, 48, and 54).

Note that the coating waste batch size shown in Table 1 is based on 6,600-lbs uranium, but that the metal waste dissolution batch size is based on 2,200-lbs uranium. These sample analyses support that the 2C waste contained less than 0.1 percent of the fission products.

3.1.1 221-B and 221-T Cell Drainage Waste

During the operation of the 221-B and 221-T Bismuth Phosphate plants, failure of process equipment, cooling jackets on process vessels, and piping occurred periodically, resulting in the discharge of cooling water, chemical solutions, and process solutions (e.g., MW, 1C, 2C wastes and plutonium product solutions) to the process cells. Each of the 40 process cells in the 221-B and 221-T Plants contained a sump that was equipped with a conductivity probe beginning in August 1946 to detect a liquid leak in the process cell (HW-7-4739-DEL, page 21). The sumps gravity drained to a 24-inch diameter vitrified clay pipe that traversed under each cell and discharged to a deep, open top, stainless steel tank, number 5-7 in section 5 (cell 10) (HW-10475-C, page 914).

Cell drainage collected in tank 5-7 was jetted to tank 5-6 or tank 5-9, which were used for sampling and chemical treatment of the cell drainage solution. Waste in tanks 5-6 and 5-9 could be jetted between these two tanks. High activity waste collected in 221-T Plant and 221-B Plant tanks 5-9 could be jetted to single-shell tank 241-T-107 and 241-B-107, respectively (HW-10475-C, page 918). Alternatively, the waste could be transferred to process vessels with the 221-T (or 221-B) Plant and processed to recover plutonium. An example of this practice is cited in the January 1948 monthly report for the Hanford Works (HW-8931-Del, page 28).

The T-Plant stack drainage waste was also collected as part of the cell drainage until May 28, 1951, after which the stack drainage was routed to the cascade of single-shell tank 241-TX-113, 241-TX-114, and 241-TX-115 (HW-21260-DEL, page 58). Also, the dissolvers located in 221-B and 221-T Plant cells 5, 6 and 7 were equipped with off-gas scrubber towers in May 1948 (HAN-45807, pages 57). The dissolver off-gas scrubbers used water to adsorb iodine and remove particulates from the dissolver off-gases. The spent scrubber solution was combined with the low-activity cell drainage waste collected in tank 5-6 (HW-10728). The dissolver off-gas scrubbers were replaced with silver chemical reactors, thus eliminating the spent scrubber solution. The first silver reactor was installed in the 221-B Plant in October 24, 1950 (HW-19898 and HW-19325, page 52) and the remaining silver chemical reactors were installed in the 221-B and 221-T Plants by January 1951 (HW-20161, page 52 and HW-21826).

Waste collected in tank 5-6 was transferred to reverse well number 216-T-3 from January 1945 through August 1946. Crib number 216-T-6 was used to dispose of the cell drainage waste from August 1946 through June 1951. After June 1951, cell drainage waste was transferred to the cascade of tanks 241-T-110, 241-T-111, and 241-T-112 (HW-55176, part V). The quantity and composition of the cell drainage solutions discharged from tank 5-6 varied (see HW-20583, page 4 and HW-33591, page 25). Table 4 provides analyses of cell drainage waste that was collected in tank 5-6 and transferred to either crib 216-T-6 or to the cascade of tanks 241-T-110, 241-T-111, and 241-T-112. As evident from the analyses provided in Table 4, the neutralized, low activity cell drainage waste contained soluble beta emitting radionuclides and plutonium.

3.2 224-B AND 224-T CONCENTRATION BUILDINGS

The process steps executed in the 224 buildings were as follows:

- The starting batch size received from the 221 buildings was 330 gallons.
- Plutonium solution from the 221 buildings was oxidized with sodium bismuthate to convert the plutonium to the +6 valence state.
- Phosphoric acid was added to produce a bismuth phosphate ($BiPO_4$) precipitate, with the plutonium still in solution. At this point, operators wanted to get rid of all the $BiPO_4$.
- The solution and precipitate were separated by centrifugation.
- Nitric acid was added to dissolve the $BiPO_4$ precipitate, with this solution removed as waste.
- Potassium permanganate ($KMnO_4$) was added to the plutonium solution to ensure all the plutonium was in the +6 valence state.
- Hydrogen fluoride and lanthanum salts were added to the plutonium solution producing a lanthanum fluoride precipitate. Fission products were carried with the lanthanum. This precipitate contained all the lanthanides (cerium, lanthanum, etc.) and residual ruthenium, samarium, europium, americium, and curium that the $BiPO_4$ could not carry out of the stream.
- The lanthanum fluoride precipitate was dissolved in nitric acid, neutralized with sodium hydroxide, and sent to waste storage tanks.
- Oxalic acid was added to the plutonium solution collected from the lanthanum fluoride precipitation step to reduce the plutonium to the +4 valence-state.
- Hydrogen fluoride and lanthanum salts were added to the plutonium solution producing a lanthanum fluoride and plutonium fluoride precipitate. The precipitate was centrifuged to collect the solids.
- Potassium hydroxide was added to convert the plutonium fluoride / lanthanum fluoride precipitate into lanthanum hydroxide and plutonium hydroxide solids.
- After centrifuging to separate the lanthanum hydroxide and plutonium hydroxide solids, these solids are reacted with nitric acid solution to dissolve the lanthanum and plutonium. The plutonium nitrate / lanthanum nitrate solution product was now ready for transfer to the 231-Z building or 234-5 building.

By this time, each original 330-gallon batch of plutonium-bearing solution that had entered the 224 Buildings was concentrated down to eight gallons. The liquid waste (designated as 224) from the lanthanum fluoride precipitation process was neutralized and transferred to the single-shell underground storage tanks. The resulting purified plutonium material was transferred to the 231-Z building and subsequently to the 234-5 building (Z Plant) beginning in 1949 for further processing.

Table 2. Analyses of Bismuth Phosphate Process Supernatants Stored [1,2]

Waste Type	Tank	pH	Pu µg/liter	Gross Beta millicuries/liter	Gross Gamma millicuries/liter	Date Sampled
Metal Waste	T-101	10.1	70	200[5]	70[5]	12-12-1946
Metal Waste	T-101	10	35	110[5]	25[5]	7-01-1947
Metal Waste	T-102	9.9	60	120	20	7-01-1947
Metal Waste	T-103	9.8	60	150	20	7-01-1947
1C/CW	B-109	9.9	40	0.65	0.28	3-18-1947
1C/CW	C-112	9.9	12	12	4.4	3-18-1947
2C	B-111	6.9	7.2E-02	2.0E-03	3.0E-03	7-1-1947
2C	B-112	6.8	4.32E?? [3]	1.5E-03	3.0E-03	7-1-1947
Waste Type	**Tank**	**pH**	**Pu µg/liter**	**Gross Beta Counts / minute/ cc**	**Gross Gamma Counts / minute/ cc**	**Date Sampled**
2C	T-110	Not reported [4]	15	4.9E+04	30	7-13-1945
2C	T-110	9.8[4]	19	6.9E+04	55	7-25-1945
2C	B-110	9.6[4]	8.5	7.0E+04	55	7-25-1945

Notes:

[1] See HW-10728 and HW-3-3220.

[2] Solids formed in each of wastes, settling to the bottom of each tanks. These sample analyses are for the supernatant only and are not representative of the sludges.

[3] The reported Pu sample analyses for tank B-112 seems to be in error and lacking an exponent in HW-10728.

[4] Prior to October 1945, the 2C waste was neutralized to a pH of approximately 10. The waste collected in tanks 241-T-110, 241-T-111, and 241-T-112 were neutralized to about pH 7 after October 1945 to precipitate bismuth and plutonium (HW-3-3220, page 13).

[5] Decreases in gross beta and gross gamma concentrations shown for the T-101 waste samples are due to decay of fission products with short half-lives.

Table 3. Analyses of Metal Waste and First Decontamination Cycle / Coating Waste Supernatant

Tank	Date Filled	Pu µg/cc	Gross Beta µCi/cc	Gross Gamma µCi/cc	Sr µCi/cc	Cs µCi/cc	Ru µCi/cc	Rare Earths + Y – Ce µCi/cc	Ce µCi/cc	Nb µCi/cc	Zr µCi/cc	Te µCi/cc
Analyses of Metal Waste Supernatant Following Uranium Extraction [1]												
C-106	Not specified				0.44	54.2						
BX-108	Not specified				0.26	132.4						
BX-109	Not specified				1.08	56.3						
C-112	Not specified				1.20	25.8						
C-109	Not specified				0.46	40.7						
C-111	Not specified				0.10	34.5						
Average Concentrations for Metal Waste					**0.59**	**57.3**						
Analyses of First Decontamination Cycle (1C) Waste Mixed with Coating Removal Waste (CW) [2]												
B-107	8-1945	1.7E-02	0.135	0.055	0.011	0.10						
T-107	9-1945	1.5E-03	0.170	0.093	0.0013	0.20						
B-108	12-1945	2.0E-02	0.183	0.044	0.022	0.12						
T-108 (Top)	12-1945	2.0E-02	0.25	0.073	0.012	0.17	0.0066	0.047	0.007	0.0018	0	1.2E-05
T-108 (Bottom)	12-1945	2.0E-02	0.25	0.070	0.012	Not reported	0.0065	0.029	0.0066	0.0024	0	3E-05
T-109	3-1946	2.6E-03	0.14	0.082	0.00038	0.15						
B-109	4-1946	1.8E-02	0.16	0.051	0.01	0.11						
T-104 (Top)	7-1946	3E-03	0.51	0.130	0.00013	0.13	0.058	0.004	0.051	0.028	0.010	2.4E-05
T-104 (Bottom)	7-1946	3E-03	0.52	0.160	0.00037	Not reported	0.059	0.003	0.050	0.028	0.015	3.6E-05
C-110	8-1946	2E-03	0.14	0.0067	0.00026	0.11						
C-111	11-1946	4.2E-03	0.16	0.069	0.01	0.13						
C-112	4-1947	3.1E-03	0.14	0.064	0.006	0.13						
U-110	4-1947	2.1E-04	0.13	0.069	0.00011	0.17						
U-111	10-1947	3.4E-04	0.12	0.060	0.00023	0.14						
TX-109 [3]	9-1949	2.7E-05	2.8	2.2	0.00087	0.27	0.34	0.0085	0.0035	0.34	1.2	8E-05
Average Concentrations for 1C / CW		**7.67E-03**	**0.39**	**0.22**	**0.0058**	**0.15**						

Notes:

[1] HW-36717, *Decontamination of Uranium Recovery Process Stored Wastes Interim Report*, May 16, 1955, W. W. Schulz, General Electric Company, Richland, Washington.

[2] HW-20195, *Radioactive Content of Stored Bismuth Phosphate First Cycle Waste Supernatants*, February 5, 1951, General Electric Company, Richland, Washington.

[3] Tank TX-109 exhibits higher gross beta and gross gamma radioactivity since this tank was sampled shortly after filling and the short-lived fission products (e.g., Ru, Nb, and Zr) had not decayed appreciably.

Table 4. Composition of Tank 5-6 Cell Drainage Waste from 221-T Plant

Year	Month	Liters	Pu Grams	Total Beta Activity Curies	Comment
Tank 5-6 Cell Drainage Transferred to 216-T-6 Crib [1,2]					
1948	January	839,900	49	88	Total beta activity does not include radioactive iodine. Samples were measured for total alpha activity. Calculated Pu mass assumes that all alpha activity measured in samples was Pu. Uranium activity in samples contributed less than 8% of the total alpha activity [1].
	February	724,461	8	73	
	March	586,188	3	789	
	April	842,778	9	461	
	May	918,007	5	72	
	June	971,810	9	295	
	July	1,057,015	6	130	
	August	831,662	4	248	
	September	857,327	5	361	
	October	830,083	4	116	
	November	980,411	6	214	
No records could be located for December 1948 through August 1949.					
1949	September	260,000	32	365	
	October	360,000	41	2800	
	November	340,000	38.2	333	
	December	430,000	48	250	
1950	January	410,000	44	210	
	February	330,000	28.5	No data reported	
	March	370,000	35	No data reported	
	April	450,000	35.6	294	
	May	370,000	33.9	363	
	June	430,000	36.6	2142	
	July	520,000	43.6	600	
	August	590,000	44.9	741	
	September	480,000	42.3	850	
	October	620,000	47.3	858	
	November	540,000	50.9	600	
	December	590,000	42.1	850	
No records could be located for January 1951 through December 1951. Beginning in June 1951, Tank 5-6 cell drainage waste along with 2C waste was routed to the cascade of tanks 241-T-110, 241-T-111, and 241-T-112.					

Table 4. Composition of Tank 5-6 Cell Drainage Waste from 221-T Plant

Year	Month	Liters	Pu Grams	Total Beta Activity Curies	Comment
Tank 5-6 Cell Drainage Waste Discharged to the Cascade of Tanks 241-T-110, 241-T-111, and 241-T-112 [3,4]					
1952	January	595,000	5.2	440	
	February	498,000	6.9	850	
	March	643,000	8.2	920	
	April	623,000	8.8	660	
	May	318,000	1.8	84	
	June	392,000	3.0	97	
	July	600,000	4.1	160	Beginning in July 1952, 224 building waste, along with tank 5-6 cell drainage and 2C wastes were routed to the cascade of tanks 241-T-110, 241-T-111, and 241-T-112. Values reported are for tank 5-6 cell drainage waste only.
	August	670,000	6.5	265	
	September	260,000	1.9	675	
	October	430,000	3.0	310	
	November	490,000	2.7	95	
	December	540,000	3.3	240	
1953	January	490,000	2.4	130	
	February	530,000	3.9	480	
	March	660,000	5.0	245	
	April	390,000	2.0	180	
	May	490,000	1.8	220	
	June	660,000	3.5	590	
	July	280,000	0.9	65	
	August	490,000	2.4	100	
	September	560,000	7.8	195	
	October	560,000	6.8	1,840	
	November	710,000	8.7	1,085	
	December	740,000	8.8	885	
1954	January	830,000	10.4	1,680	
	February	820,000	14.2	16,420	
	March	860,000	18.6	5,305	
	April	540,000	8.4	2,175	
	May	790,000	10.6	1,760	
	June	810,000	9.5	2,390	
	July	1,030,000			Radionuclide content not reported separately for 5-6 Cell drainage waste from July 1954 thru June 1955 (HW-38562, page 26).
	August	1,150,000			
	September	1,090,000			
	October	800,000			
	November	730,000			
	December	1,100,000			
1955	January	1,370,000			
	February	950,000			
	March	1,460,000			
	April	1,380,000			

Table 4. Composition of Tank 5-6 Cell Drainage Waste from 221-T Plant

Year	Month	Liters	Pu Grams	Total Beta Activity Curies	Comment
	May	1,410,000			
	June	1,440,000			
The volume and radionuclide content of tank 5-6 cell drainage waste were not recorded separate from other wastes transferred into the cascade of tanks 241-T-110, 241-T-111, and 241-T-112 after July 1954.					

Notes:

(1) HW-11908
(2) HW-20583
(3) HW-25301
(4) HW-33591
(5) Analyses of the combined 2C / 224 building / tank 5-6 waste supernatant stored in tank 241-T-112 conducted on August 6, 1952 and September 24, 1952 indicate that the total beta emitters was comprised of 35 to 50% ruthenium, 35 to 50% cesium, 4 to 8% cerium, yttrium, and other rare earths, and 6 to 11% undetermined (HW-27035, page 8).

3.3 221-B PLANT FISSION PRODUCTS PROCESSING

From August 1963 through June 1966, B-Plant was used in conjunction with the PUREX facility, 244-CR Vault, and the 201-C Hot Semiworks (renamed Strontium Semiworks in 1963) to separate strontium-90 and rare earths (i.e., cerium-144 and promethium-147) from high-level waste solutions. Then, from July 1966 through December 1967, equipment was replaced within B-Plant to expand the processing capability to include cesium removal from fission high-level waste solutions using ion exchange equipment. The strontium and rare earths processing equipment was also replaced to include only strontium removal using a solvent extraction equipment, followed by precipitation and centrifugation equipment for purifying the strontium. Each of the fission products processing events in the B-Plant is discussed in more detail in the following sections.

3.3.1 STRONTIUM AND RARE EARTHS PROCESSING

On September 18, 1961 (HW-71187-DEL, page F-2), renovation of cells 5 through 12 within B-Plant canyon was initiated to use these cells for separating strontium and rare earths from a mixed fission product solution (HW-69011). Construction activities were completed, and the facility was accepted by operations on January 31, 1963 (HW-76848-DEL, page B-2). Processing of radioactive waste in cells 5 through 12 at the B-Plant commenced on August 2, 1963 (HW-78817-DEL, pages B-2 and G-2).

B-Plant was used in conjunction with the PUREX facility, 244-CR Vault and the 201-C Hot Semiworks to separate strontium-90, cerium-144 and promethium-147 from high-level waste solutions. The PUREX facility generated a first cycle raffinate solution from the solvent extraction reprocessing of irradiated reactor fuel (i.e., high-level waste). The first cycle raffinate solution was highly acidic and contained most of the fission products (e.g., strontium-89/90, cerium-144, promethium-147, and cesium-137) that were separated from the uranium and plutonium during the reprocessing of irradiated reactor fuel. The acidity of the first cycle

raffinate solution was reduced by addition of sugar and digestion at elevated temperature to decompose the nitric acid solution.

In a section of the PUREX facility known as the head-end, first cycle raffinate solution was reacted with sodium sulfate and lead nitrate to precipitate strontium and rare earth (i.e., cerium and promethium) fission products (HW-63051 and HW-69534). Lead co-precipitated with strontium and increased the amount of strontium precipitated from the first cycle raffinate solution. The resulting strontium and rare earth precipitate was centrifuged and washed to separate the supernatant, which contained soluble fission products such as cesium-137, zirconium-niobium-95, and ruthenium-rhodium-106. The supernatant containing the soluble fission products (e.g., cesium-137, zirconium-niobium-95, and ruthenium-rhodium-106) was neutralized and transferred to underground storage tanks. The strontium and rare earth precipitate was metathesized to soluble carbonates by addition of sodium carbonate. The strontium and rare earth carbonate precipitates were then dissolved in nitric acid and transferred to B-Plant via 244-CR Vault for further processing.

In B-Plant, the strontium nitrate / rare earth nitrate solution were processed to form separate solutions containing strontium and rare earths (HW-77016). The strontium nitrate / rare earth nitrate solution was reacted with oxalic acid to precipitate the rare earths along with lead, leaving strontium in solution. The precipitate was centrifuged to separate the strontium solution from the rare earth precipitate. The strontium solution was stored in B-Plant and transferred periodically to the 201-C Hot Semiworks for purification. The rare earth precipitate was dissolved in nitric acid and stored in B-Plant for further processing.

Lead was removed from the rare earth solution by adding sodium hydroxide solution to form soluble plumbite and insoluble rare earth hydroxide precipitates (HW-81373, RL-SEP-197, page G-2, and HAN-90907, page 21). The plumbite was separated from the rare earth hydroxide precipitate by centrifugation and discarded to the single-shell tanks. The rare earth hydroxide precipitate was washed with sodium hydroxide solution to remove soluble lead and the wash solution was also discarded to the single-shell tanks. The rare earth hydroxide precipitate was dissolved in nitric acid, stored in B-Plant, and eventually transferred to the 201-C Hot Semiworks for purification.

Processing of strontium and rare earth solutions within B-Plant continued until June 1966 (HAN-95105-DEL, page 15). Separations of strontium and rare earths from the first cycle raffinate solution continued to be conducted in the head-end section of the PUREX facility through February 8, 1967 (HAN-96805-DEL, page AIII-4). The strontium and rare earth solution was transferred from PUREX to the 244-CR Vault for storage from July 1966 through February 1967, while equipment modifications were conducted at B-Plant.

3.3.2 CESIUM AND STRONTIUM PROCESSING

From July 1966 (HAN-95284-DEL, page 13) through October 1967 (HAN-98918-DEL, page AIII-2), equipment within the 221-B Plant was flushed and replaced with new equipment for separating cesium and strontium from high-level waste. In January 1967 (HAN-96590-DEL, page AIII-4) and in March 1967 (HAN-97066-DEL, page AIII-4), testing was conducted of a new centrifuge and a precipitation-decantation-centrifugation technique for separating iron and aluminum from PUREX sludge waste. Construction activities continued to be conducted in the 221-B Plant throughout 1967.

On December 27, 1967 (HAN-99396-DEL, page AIII-3), alkaline supernatants stored in the single-shell tanks were transferred to B-Plant, and cesium was separated using an ion exchange process. Cesium ion exchange processing continued at B-Plant until October 1983 using at first inorganic and later organic ion exchange materials (RHO-RE-SA-169). Cesium was also precipitated from acidic, PUREX high-level waste (known as CAW) using phosphotungstic acid (PTA), with the cesium precipitate dissolved in sodium hydroxide solution and processed through the ion exchange equipment for cesium recovery (ARH-CD-917). After separation of cesium, the alkaline supernatants were transferred directly to underground storage tanks. The ion exchange process used an ammonium carbonate / ammonium hydroxide solution to separate sodium from cesium on the ion exchange media. The aqueous wastes that contained ammonium were processed in the Cell 23 evaporator to concentrate these wastes and volatilize ammonia before transferred to underground storage tanks.

On January 31, 1968, the solvent extraction equipment installed in B-Plant was operated to purify the inventory of rare earth solutions stored at B-Plant (HAN-99604-DEL, page AIII-3). The semi-purified promethium - cerium solution was stored in B-Plant process tank 6-2 (HAN-100127-DEL, page AIII-3). Separation of strontium from the strontium and rare earths solutions stored in the 244-CR Vault was then conducted in March 1968 using the solvent extraction equipment (HAN-100127-DEL, page AIII-3).

The B-Plant solvent extraction equipment began processing the PUREX first cycle raffinate solution to separate strontium on April 20, 1968 (HAN-100357-DEL, page AIII-3). The processing of PUREX first cycle raffinate solution was completed on August 30, 1968 (PR-REPORT-SEP68-DEL, page AIII-3). The B-Plant solvent extraction equipment was then used to separate strontium from PUREX high-level waste sludges. The PUREX high-level waste sludges were dissolved in nitric acid (known as PAS) in the 244-AR Vault and transferred to B-Plant for centrifugation to separate solids. The clarified solution was process in the solvent extraction equipment to separate strontium (PR-REPORT-SEP-68-DEL, page AIII-4). In addition, the B-Plant solvent extraction equipment was operated periodically to separate strontium from CAW solutions following the PTA processing to separate cesium. Strontium separation from high-level waste solutions using the solvent extraction equipment continued at B-Plant until 1977. The aqueous waste from the solvent extraction process was evaporated in the Cell 23 evaporator and transferred to underground storage tanks.

4.0 TRANSURANIC ANALYSES OF WASTE IN TANKS 241-T-110, 241-T-111 AND 241-T-112

The Hanford Site prepares a Best Basis Inventory (BBI) estimate of the composition of the wastes stored in all 177 Hanford Site underground storage tanks. The BBI effort involves developing and maintaining waste tank inventories comprising 25 chemical and 46 radionuclide components in the 177 Hanford Site underground storage tanks. Waste sample analyses, process knowledge, and waste templates are used to create the BBIs. These BBIs provide waste composition data necessary as part of the River Protection Project (RPP) process flowsheet modeling work, safety analyses, risk assessments, and system design for retrieval, treatment, and disposal operations. Development and maintenance of the BBI is an on-going effort, with the current BBIs available electronically through TWINS, http://twins.pnl.gov/data/datamenu.htm.

The BBI for the tank 241-T-110 waste is based on the analyses of two core samples obtained in 1996. Composites of these core samples were analyzed for non-radioactive components and total alpha concentrations. A re-analysis of a composite sample was conducted in 2003 to determine the concentrations cesium-137, strontium-90 and individual transuranic elements with half-life greater than 20-years (i.e. neptunium-237, plutonium-238, plutonium-239, plutonium-240 and americium-241). The analytical results were reviewed and used along with engineering judgment to determine the best basis inventory for the waste stored in tank 241-T-110. The mean, total alpha analysis for the waste stored in tank 241-T-110 is 53ηCi/g. The uncertainty estimates for the total alpha analyses for the waste stored in tank 241-T-110 were evaluated (RPP-10983). The upper 95% confidence limit for the gross alpha analyses of the waste stored in tank 241-T-110 is 62ηCi/g. The sum of the neptunium-237, plutonium-238, plutonium-239, plutonium-240 and americium-241 concentrations analyzed in the composite core sample is approximately 83.3ηCi/g, as reported on October 11, 2004 from the Tank Waste Information Network (TWINS) database; http://twins.pnl.gov/. These analyses indicate that the concentration of alpha-emitting transuranic isotopes with half-life greater than 20 years is less than 100ηCi/g in the waste stored in tank 241-T-110.

The BBI for the tank 241-T-111 waste is based on two core samples obtained in 1991. Composite of these core samples were analyzed for non-radioactive components, select radionuclides, total alpha and transuranic element concentrations. The analytical results were reviewed and used along with engineering judgment to determine the best basis inventory for the waste stored in tank 241-T-111. The mean total alpha analyses and lower 95% confidence limit for the waste stored in tank 241-T-111 are 371ηCi/g and 289ηCi/g (7G300-02-JGF-009). The total alpha analyses of the waste in tank 241-T-111 are support by analyses of this waste for neptunium-237, plutonium-238, plutonium-240, plutonium-240 and americium-241. The sum of these transuranic elements is approximately 186.5ηCi/g in the waste stored in tank 241-T-111, as reported on October 11, 2004 from the Tank Waste Information Network (TWINS) database; http://twins.pnl.gov/.

The BBI for the tank 241-T-112 waste is based on two core samples obtained in 1997. These two core samples were analyzed to determine gross alpha and non-radioactive constituents in the liquid and solids portions of these samples. Template values were used for constituents below

the detection limits for sample data or constituents not measured from the sampling event. Templates are based on sampling data from tanks that contain the same waste type as tank 241-T-112, supplemented with Revision 5 of the HDW model data (RPP-19822). The mean, total alpha analysis for the sludge fraction of the waste stored in tank 241-T-112 is 255ηCi/g. The template based sum of neptunium-237, plutonium-238, plutonium-240, plutonium-240 and americium-241 concentrations in the sludge fraction of the waste stored in tank 241-T-112 is approximately 255.2ηCi/g.

5.0 SUMMARY

Tanks 241-T-110 received 2C waste from reprocessing of spent nuclear fuel at the 221-T Plant from January 1945 through December 1954, low-activity cell drainage waste from June 1951 through December 1954, and 224 wastes from May 1952 through December 1954. Tank 241-T-111 received 2C waste from the 221-T Plant from January 1945 through October 1956, low-activity cell drainage waste from June 1951 through October 1956, 224 wastes from May 1952 through October 1956, and equipment decontamination waste from December 1959 through June 1967. Tank 241-T-112 continued to receive 221-T Plant equipment decontamination waste until June 1973. Tank 241-T-112 also received a mixture of coating removal waste 221-B Plant cesium ion exchange process waste from tank 241-T-106 in March 1973.

The concentrations of the transuranic elements (i.e. sum of neptunium-237, plutonium-238, plutonium-240, plutonium-240 and americium-241) in the waste stored in tanks 241-T-110, 241-T-111 and 241-T-112 (sludge only) are approximately 83.3ηCi/g, 186.5ηCi/g and 255.2ηCi/g, as reported on October 11, 2004 from the TWINS database.

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APPENDIX A

VOLUME OF SOLIDS AND TOTAL WASTE IN TANKS 241-T-110, 241-T-111, AND 241-T-112

January 1945 through December 1975

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Reference	Year	Period	Cascade of Tanks T-110, T-111, and T-112 Percent Filled						Comments from Reference Document
HW-7-1293-DEL	1945	Jan	Not reported						
HW-7-1388-DEL, page 18		Feb	7.8%						
HW-7-1544-DEL, page 21		March	9.7%						
HW-7-1649-DEL, page 20		April	12.0%						
HW-7-1793-DEL, page 22		May	15.4%						
HW-7-1981-DEL, page 23		June	17.6%						
HW-7-2177-DEL, page 22		July	21.2%						
HW-7-2361-DEL, page 21		August	26.4%						
HW-7-2548-DEL, page 22		September	32.6%						"Technical Department work has demonstrated that the precipitate formed on neutralization will carry more residual product into the sludge at pH of 7 than at the pH of 9 to 10 previously used. Consequently, the second cycle wastes are now being neutralized to a pH of about 7. This may permit disposal of the supernatant solution through dry-wells at some future date ..."
HW-7-2706-DEL, page 21		October	38.7%						"The second decontamination cycle waste in tank (110) in the T Plant waste storage area was filled and began to cascade into the second tank of the series on about October 15."
HW-7-2957-DEL, page 21		November	49.0%						
HW-7-3171-DEL, page 21		December	58.5%						
HW-7-3378-DEL, page 24	1946	Jan	67.1%						
HW-7-3566-DEL, page 21		Feb	74.3%						
HW-7-3751-DEL, page 21		March	78.4%						
HW-7-4004-DEL, page 21		April	86.1%						
HW-7-4193-DEL, page 21		May	91.3%						

Reference	Year	Period	Cascade of Tanks T-110, T-111, and T-112 Percent Filled						Comments from Reference Document
HW-7-4343-DEL, page 23		June	97.5%						
HW-7-4542-DEL, page 21-22		July	100.0%						"The installation of a new underground line from diversion box 153 in the T Waste Area to the second cascade tank in the X104 series permitted the diversion, on 7-22-46, of second cycle waste and stack drainage from T Plant to the remaining two tanks in this series."
HW-7-4739-DEL, page 23		August	100.0%						2C waste transferred from 221-T into T-105, which cascades to T-106.
HW-7-5194-DEL, page 26		September	100.0%						
HW-7-5362-DEL, page 27-28		October	100.0%						"Tanks X-110-T, X-111-T and X-112-T in T Plant were checked by means of ionization chambers and it was determined that while the sludge level in the first tank was evenly distributed on the bottom of the tank to a depth of 38", there was no indication of sludge in the second and third tanks in the series." 38" of sludge corresponds to ~84,030-gallons.
HW-7-5505-DEL, page 28		November	100.0%						
HW-7-5630-DEL, page 25		December	100.0%						
HW-7-5802-DEL, page 26	1947	Jan	100.0%						
HW-7-5944-DEL, page 25		Feb	100.0%						"Excavation for an underground crib system and tile file adjacent to the 241 T Waste Storage Tank Farm in T Plant for the proposed handling of second cycle waste supernatants was started in February."
HW-7-6048-DEL, page 24		March	100.0%						
HW-7-6184-DEL, page 26		April	100.0%						
HW-7-6391-DEL, page 24		May	100.0%						
HW-7-7454-DEL, page 26		June	100.0%						
HEW-7283-DEL, page 26		July	100.0%						
HEW-7504-DEL, page 27		August	100.0%						
HW-7795-DEL, page 26-27		September	71.0%						"In T Plant, one tank, X-111-T, containing second cycle waste was disposed of to the recently installed underground crib system (Project C-120)."

Reference	Year	Period	Cascade of Tanks T-110, T-111, and T-112 Percent Filled						Comments from Reference Document
HW-7997-DEL, page 27		October	67.0%						2C waste transferred from 221-T into T-105, which cascades to T-106.
HW-8267-DEL, page 29		November	67.0%						"In order to safely complete the tie-in of the two lines from the 153-T diversion box to the X-111T and X-112T underground waste storage tanks, 20,000-gallons of second cycle waste were jetted from X-112T tank to the second cycle waste crib in T Plant. No further jetting of second cycle waste will be done until the perforating device mentioned in last month's report is available."
HW-8438-DEL, page 27		December	67.0%						2C waste transferred from 221-T into T-105, which cascades to T-106.
HW-8931-DEL, page 28	1948	Jan	67.0%						2C waste transferred from 221-T into T-105, which cascades to T-106.
HW-9191-DEL, page 30		Feb	67.0%						2C waste transferred from 221-T into T-105, which cascades to T-106.
HW-9595-DEL, page 32		March	70.8%						Tanks T-105 and T-106 are filled with 2C waste. 2C waste transferred from 221-T into cascade of T-110, T-111, and T-112.
HW-9922-DEL, page 31-32		April	75.1%						~360,000-gallons of 2C waste jetted from T-105 to crib.
HW-10166-DEL, page 33		May	79.3%						
HW-10378-DEL, page 30		June	84.1%						
HW-10714-DEL, page 32-33		July	89.0%						2C waste jetted from tank T-106 to crib.
HW-10993-DEL, page 35-36		August	67.5%						"The cribbing of tank X-112-T was started on August 4, 1948 and was nearing completion at month end."
HW-11226-DEL, page 32-33		September	66.0%						"At T Plant the cribbing of the X-112-T tank was discontinued after approximately 450,000 gallons of waste had been jetted to the crib to permit installation of an experimental sand filter for the purpose of determining the feasibility of removing the activity from the waste supernatant by this method prior to discharging it to the ground."
HW-11499-DEL, page 33-34		October	66.8%						"At T Plant, the cribbing of second cycle wastes in tank X-112-T was completed."
HW-11835-DEL, page 36		November	71.0%						
HW-12086-DEL, page 38		December	77.3%						

Reference	Year	Period	Cascade of Tanks T-110, T-111, and T-112 Percent Filled						Comments from Reference Document
HW-12391-DEL, page 39	1949	Jan	84.4%						
HW-12666-DEL, page 35		Feb	89.7%						
HW-12937-DEL, page 40-41		March	87.2%						"The cribbing of second cycle waste supernatant from X-112-T tank was started on March 17, 1949. At month end 151,250-gallons of this material has been jetted to the crib. Average age of material being cribbed is estimated to be nine months. Analysis of a sample of the waste being cribbed is as follows: AT 100 c/m/ml, Beta's 1730 c/m/ml, Gamma's 10 c/m/ml, pH 7.1, suspended solids 0.1%, Uranium nil.
HW-13190-DEL, page 39-40		April	71.0%						"The cribbing of second cycle waste from X-112-T tank which was started on March 17, 1949, was continued with a total of 319,000-gallons being jetted to the crib during this month."
HW-13561-DEL, page 41-41		May	71.9%						"The cribbing of second cycle waste from X-112-T tank, which was started in March, was completed with 55,000-gallons being cribbed during this month."
HW-13793-DEL, page 41		June	87.2%						Reported percent fill may be error.
HW-14043-DEL, page 43		July	77.6%						
HW-14338-DEL, page 44		August	81.9%						
HW-14596-DEL, page 43		September	86.2%						
HW-14916-DEL, page 44		October	92.4%						
HW-15267-DEL, page 43-45		November	82.6%						""In order to provide additional space for the storage and settling of second decontamination cycle wastes in the X-110-T series of tanks in the 241-T area, cribbing of supernatant from X-112-T (final tank in 110-T series) was resumed on November 18 ...". "Cribbing of 215,000-gallons of supernatant was completed during the month."
HW-1550-DEL, page 42-43		December	69.4%						"Cribbing of 578,000-gallons of second cycle waste from X-112-T tank in the 200 West Area was concluded on December 19, for a total of 2,955,000-gallons of this type waste cribbed in the 200 West Area since cribbing of second cycle supernatant was first initiated."
HW-15843-DEL, page 45	1950	Jan	76.2%						
HW-17056-DEL, page 45		Feb	81.8%						
HW-17410-DEL, page 47-49		March	89.9%						"A total of 27,300-gallons of second cycle decontamination waste having its origin from the bismuth phosphate process semi-works, which was formerly operated in the 300 Area, was placed in the X-112-T waste storage tank during this month."

Reference	Year	Period	Cascade of Tanks T-110, T-111, and T-112 Percent Filled						Comments from Reference Document
HW-17660-DEL, page 46-47		April	92.0%						"Cribbing of supernatant from X-112-T second cycle waste storage tank in the 200 West Area was started late in the month when this tank became 85% filled".
HW-17971-DEL, page 44-45		May	71.9%						"In 200 West Area, the cribbing of supernatant from X-112-T second cycle waste storage tank, started late in April, has totaled 462,000-gallons to date".
HW-18221-DEL, page 43-45		June	70.8%						"Approximately 129,000-gallons of second cycle waste supernatant were cribbed during the month from the 112-T tank in 200 West Area".
HW-18473-DEL, page 46		July	79.5%						
HW-18740-DEL, page 50		August	89.4%						
HW-19021-DEL, page 49		September	97.2%						

Reference	Year	Period	Cascade of Tanks T-110, T-111, and T-112 Waste Volume (x 1000 gallons)						Comments from Reference Document
HW-19325-DEL, page 49-50		October	1382						"With the X-112-B and X-112-T tanks each becoming half full during the month, cribbing of the second decontamination cycle waste from each was resumed. A total of 327,250-gallons in T Plant and 507,700 gallons in B Plant had been cribbed by month end without incident."
HW-19622-DEL, page 49		November	1154						"A total of 349,250-gallons of second decontamination cycle waste supernatant was cribbed from X-112-T tank in the 200 West Area during the month."
HW-19842-DEL, page 50-51		December	1073						"Disposal of second decontamination cycle waste to underground cribs was made in December as tabulated below: 200 East Area from tank X-112-B: 249,000-gallons 200 West Area from tank X-112-T: 118,200-gallons
HW-20161-DEL, page 50	1951	Jan	1297						
HW-20438-DEL, page 50		Feb	1448						"Cribbed as necessary."
HW-20671-DEL, page 54		March	1467						"143,000-gallons cribbed from 112-T."
HW-20991-DEL, page 52-53		April	1596						"Cribbed when necessary." "The tie-line from tank 241-T-112 to the 241-T crib was completed during the month on Project C-415, which will allow constant overflow of settled second decontamination cycle waste in conjunction with Section 5 waste." [Section 5 waste is low-activity cell drainage from T Plant tank 5-6]
HW-21250-DEL, page 56-57		May	1629						"185,600-gallons estimated cribbed by cascade." "The settled second cycle decontamination cycle waste started to overflow constantly from the 241-T-112 tank on May 8, 1951 and has continued uneventfully since that time. It is expected that Section 5 waste effluents, originating in the Canyon Buildings, will be combined with second cycle decontamination wastes at both B and T Plants, during June."
HW-21506-DEL, page 55		June	1629						"Cascades to Crib". "... parallel changes were completed in both the 200 East and West Areas, whereby the Section 5 waste effluents, originating in the Canyon Buildings, were combined with the second decontamination cycle waste facilities, rather than being discharged directly to underground cribs."
HW-21802-DEL, page 41		July	1629						"Cascades to Crib".
HW-22075-DEL		August	Not reported						
HW-22304-DEL		September	Not reported						
HW-22610-DEL		October	Not reported						
HW-22875-DEL		November	Not reported						
HW-23140-DEL		December	Not reported						

Reference	Year	Period	T-110	T-110	T-111	T-111	T-112	T-112	Comments from Reference Document
			Waste Volume (x 1000 gallons)						
			Total	Sludge	Total	Sludge	Total	Sludge	
no report	1952	Jan							Hanford Site monthly reports only list percent filled for cascade prior to 1952.
HW-27897		Feb							No information on individual tanks
HW-27898		March	530	Not reported	530	Not reported	569	Not reported	T-110, T-111, T-112 operate as cascade and receive 2C waste. T-112 cascades to crib.
HW-27838		April to June	530	Not reported	530	Not reported	569	Not reported	On 5-29-52, 224-T wastes were tied into 5-6 stream feeding to 110-111-112-T at diversion box 241-T-152.
HW-27839		July to Sept.	530	Not reported	530	Not reported	569	Not reported	5-6, 2C and 224 waste routed to these tanks. T-112 cascades to crib.
HW-27840		Oct. to Dec.	530	Not reported	530	Not reported	569	Not reported	
HW-27841	1953	Jan	530	530	530	191	569	28	
HW-27842		Feb	530	530	530	191	519	28	
HW-27775		March	530	530	530	191	569	29	
HW-28043		April	530	530	530	233	569	34	
HW-28377		May	530	530	539	235	569	35	
HW-28712		June	530	530	530	246	569	45	
HW-29054		July	530	530	530	252	569	50	
HW-29242		August	530	521	530	186	569	2	
HW-29624		September	530	521	530	213	569	2	
HW-29905		October	530	522	530	220	564	5	
HW-30250		November	530	523	530	229	569	6	
HW-30498		December	530	530	530	230	569	7	
HW-30851	1954	Jan	530	530	530	245	569	12	
HW-31126		Feb	530	530	530	277	569	22	
HW-31374		March	530	530	530	299	569	33	
HW-31811		April	530	530	530	299	569	33	
HW-32110		May	530	530	530	299	569	33	
HW-32389		June	530	530	530	312	569	33	
HW-32697		July	530	530	530	345	569	33	
HW-33002		August	530	530	530	355	569	33	
HW-33396		September	Illegible	Illegible	Illegible	Illegible	Illegible	Illegible	
HW-33544		October	530	530	530	377	569	33	
HW-33904		November	530	530	530	417	569	33	
HW-34412		December	530	530	530	439	569	33	T-110 full of sludge. T-111 and T-112 active 2C and 224 waste cascade to crib.
HW-35022	1955	Jan	530	530	530	464	569	33	
HW-35628		Feb	530	530	530	430	569	33	T-111 and T-112 active 2C and 224 waste cascade to crib.

Reference	Year	Period	T-110	T-110	T-111	T-111	T-112	T-112	Comments from Reference Document
			Waste Volume (x 1000 gallons)						
			Total	Sludge	Total	Sludge	Total	Sludge	
HW-36001		March	530	530	530	487	569	33	T-111 and T-112 active 2C and 224 waste cascade to crib.
HW-36553		April	530	530	398	362	322	166	T-111 pumping to T-112 at month end. 833,000-gallons supernatant pumped to open ditch.
HW-37143		May	530	530	530	387	569	170	T-112 cascades to crib.
HW-38000		June	530	530	530	402	569	170	T-112 cascades to crib.
HW-38401		July	530	530	530	408	569	170	
HW-38926		August	530	530	530	417	569	170	
HW-39216		September	530	530	530	430	569	170	
HW-39850		October	530	530	530	440	569	170	
HW-40208		November	530	530	530	465	569	170	
HW-40816		December	530	530	530	465	569	170	T-112 pumps to crib at intervals.
HW-41038	1956	Jan	530	530	530	482	442	170	T-112 pumps to crib at intervals.
HW-41812		Feb	530	530	530	499	472	170	
HW-42394		March	530	530	530	507	439	170	
HW-42993		April	530	530	530	510	441	170	
HW-43490		May	530	530	530	510	435	170	
HW-43895		June	530	530	530	510	429	170	T-111 Receives from T Plant. T-112 pumps to crib at intervals.
HW-44860		July	530	530	530	510	429	170	
HW-45140		August	530	530	530	510	429	170	
HW-45738		September	530	530	530	510	429	170	
HW-46382		October	530	530	530	Not reported	439	Not reported	
HW-47052		November	530	530	530	510	451	170	
HW-47640		December	530	530	530	510	470	170	
HW-48144	1957	Jan	481	530	543	510	475	170	T-112 estimated reading.
HW-48846		Feb	481	530	554	510	475	170	T-112 estimated reading.
HW-49523		March	481	530	560	510	422	170	T-112 latest electrode reading.
HW-50127		April	483	530	560	510	417	170	
HW-50617		May	527	530	560	510		170	HW-50617 not legible. Values from HW-83906-C-RD, page 46. Corrected T-110 reading.
HW-51348		June	527	530	560	510	448	170	
HW-51858		July	527	530	557	510	414	170	
HW-52414		August	527	46	557	510	417	170	
HW-52932		September	527	46	557	510	417	170	
HW-53573		October	527	46	557	510	417	170	
HW-54067		November	527	46	557	510	417	170	

Reference	Year	Period	T-110	T-110	T-111	T-111	T-112	T-112	Comments from Reference Document
			Waste Volume (x 1000 gallons)						
			Total	Sludge	Total	Sludge	Total	Sludge	
HW-54519		December	527	46	557	510	417	170	
HW-54916	1958	Jan	527	46	557	510	417	170	
HW-55264		Feb	527	46	557	510	420	170	
HW-55630		March	527	46	557	510	420	170	
HW-55997		April	527	46	557	510	420	170	
HW-56357		May	527	46	524	510	420	170	Corrected calculation for T-111 waste volume.
HW-56761		June	527	46	524	510	420	170	
HW-57122		July	527	46	524	510	420	170	
HW-57550		August	527	46	524	510	420	170	
HW-57711		September	527	46	527	510	420	170	
HW-58201		October	527	46	527	510	422	170	
HW-58579		November	527	46	527	510	425	170	
HW-58831		December	527	46	527	510	425	170	
HW-59204	1959	Jan	527	46	527	510	425	170	
HW-59586		Feb	527	46	524	510	439	170	
HW-60065		March	524	46	524	510	439	170	
HW-60419		April	524	46	524	510	439	170	
HW-60738		May	524	46	524	510	442	170	
HW-61095		June	524	46	524	510	444	170	
HW-61582		July	524	46	524	510	444	170	
HW-61952		August	524	46	524	510	444	170	
HW-62421		September	524	46	524	510	442	170	
HW-62723		October	524	46	524	510	442	170	
HW-63083		November	524	46	524	510	442	170	
HW-63559		December	524	46	527	510	442	170	T-111 received 2,750-gallons of waste from 221-T (HW-83906-D-RD, page 92).
HW-63896	1960	Jan	524	46	527	510	442	170	
HW-64373		Feb	524	46	527	510	447	170	
HW-64810		March	524	46	527	510	450	170	T-112 received 3,000-gallons 221-T waste.
HW-65272		April	524	46	527	510	450	170	
HW-65643		May	524	46	527	510	422	170	T-112 received 16,000-gallons from 221-T. Pumped out to TY crib 44,000-gallons.
HW-66187		June	524	46	527	510	429	170	T-112 received 7,000-gallons from 221-T.
HW-66557		July	524	46	527	510	461	170	T-112 received 32,000-gallons from 221-T.
HW-66827		August	524	46	527	510	461	170	

Reference	Year	Period	T-110	T-110	T-111	T-111	T-112	T-112	Comments from Reference Document
			Waste Volume (x 1000 gallons)						
			Total	Sludge	Total	Sludge	Total	Sludge	
HW-67696		September	524	46	527	510	461	170	
HW-67705		October	524	46	527	510	461	170	
HW-68291		November	524	46	527	510	461	170	
HW-68292		December	524	46	529	510	403	170	T-112 jetted 58,000-gallons to TY crib.
HW-71610	1961	Jan. to June	524	46	527	510	395	170	T-112 received 221-T waste and pumped to TY crib.
HW-83906-D-RD pg. 197	*1961*	*Jan. to June*	*524*	*530*	*521*	*510*	*396*	*170*	*T-111 cascades to T-112. T-112 receives 221-T crib waste and pumps to TY crib.*
HW-72625	1961	Jul to Dec.	524	46	527	510	395	170	
HW-83906-E-RD pg. 8	*1961*	*Jul to Dec.*	*524*	*530*	*527*	*510*	*395*	*170*	*T-111 receives waste from 221-T. Cascades to T-112, which is pumped to TY crib.*
HW-74647	1962	Jan. to June	524	46	524	510	442	170	221-T waste cascades from T-111 to T-112 and then pumped to cribs.
HW-76223	1962	Jul to Dec.	524	46	524	510	395	170	221-T waste cascades from T-111 to T-112 and then pumped to cribs.
HW-78279	1963	Jan. to June	524	46	524	510	395	170	221-T waste cascades from T-111 to T-112 and then pumped to cribs.
HW-80379	1963	Jul to Dec.	524	46	524	510	430	170	T-111 received wastes from 221-T cascades to T-112, which is pumped to TY crib.
HW-83308	1964	Jan. to June	524	46	524	510	430	170	221-T waste cascades from T-111 to T-112 and then pumped to cribs.
HW-83906-E-RD pg. 58	1964	Jul to Dec.	524	46	524	510	433	170	T-111 received wastes from 221-T cascades to T-112, which is pumped to TY crib.
HW-83906-E-RD pg. 64	1965	Jan. to June	530	46	541	510	442	170	T-111 received wastes from 221-T cascades to T-112, which is pumped to TY crib.
HW-83906-E-RD pg. 70	1965	July to Sept.	530	46	537	510	447	170	T-111 receives waste from 221-T, cascades to T-112, which is pumped to the TY crib.
HW-83906-E-RD pg. 78	1965	Oct. to Dec.	530	46	540	510	408	170	T-111 receives waste from 221-T. T-112 waste pumped to TY crib.
no report	1966	Jan. to March							
HW-83906-E-RD pg. 86	1966	April to June	530	46	539	510	425	170	T-111 receives waste from 221-T. Cascades to 112-T, pumped to TY crib.
ISO-538	1966	July to Sept.	532	508	538	442	392	40	221-T waste cascades from T-111 to T-112, then pumps to cribs.
ISO-674	1966	Oct. to Dec.	532	508	538	442	392	40	221-T waste cascades from T-111 to T-112, then pumps to cribs.
ISO-806	1967	Jan. to March	532	508	538	442	508	40	221-T waste cascades from T-111 to T-112, then pumps to cribs.
ISO-967	1967	April to June	534	508	538	442	362	40	Transferred 146,000-gallons from T-112 to TX-118.
									Beginning in July 1967, waste status summary reports indicate that equipment decontamination waste from 221-T Plant was routed to T-112.
ARH-95	1967	July to Sept.	534	508	538	442	73	40	289,000-gallons transferred from T-112 to TX-118
ARH-326	1967	Oct. to Dec.	534	508	540	442	371	170	T-112 Received 298,000-gallons from 221-T

Reference	Year	Period	T-110	T-110	T-111	T-111	T-112	T-112	Comments from Reference Document
			Waste Volume (x 1000 gallons)						
			Total	Sludge	Total	Sludge	Total	Sludge	
ARH-534	1968	Jan. to March	535	46	538	510	508	170	T-112 Received 141,000-gallons from 221-T
ARH-721	1968	April to June	534	508	538	442	237	40	T-112 Transferred 382,000-gallons to REDOX Evaporator. Received 111,000-gallons from 221-T.
ARH-871	1968	July to Sept.	534	508	538	442	365	40	T-112 Transferred 125,000-gallons to REDOX Evaporator. Received 253,000-gallons from 221-T.
ARH-1061	1968	Oct. to Dec.	534	508	538	447	266	24	T-112 Transferred 354,000-gallons to REDOX Evaporator. Received 255,000-gallons from 221-T.
ARH-1200 A	1969	Jan. to March	534	508	538	447	406	24	T-112 Transferred 86,000-gallons to REDOX Evaporator. Received 226,000-gallons from 221-T.
ARH-1200 B	1969	April to June	534	508	538	447	165	24	T-112 Transferred 20,000-gallons to REDOX Evaporator. Received 227,000-gallons from 221-T. Transferred 448,000 to 103-TY.
ARH-1200 C	1969	July to Sept.	534	508	538	447	354	24	T-112 received 189,000-gallons of decontamination waste from 221-T Plant.
ARH-1200 D	1969	Oct. to Dec.	534	293	537	233	174	33	T-112 received 232,000-gallons of decontamination waste from 221-T Plant. T-112 transferred 413,000-gallons to REDOX evaporators.
ARH-1666 A	1970	Jan. to March	534	293	539	233	290	33	T-112 Transferred 111,000-gallons to REDOX Evaporator. Received 228,000-gallons from 221-T.
ARH-1666 B	1970	April to June	534	293	538	233	343	32	T-112 Transferred 109,000-gallons to REDOX Evaporator. Received 161,000-gallons from 221-T.
ARH-1666 C	1970	July to Sept.	534	293	539	233	351	32	T-112 Transferred 132,000-gallons to REDOX Evaporator. Received 140,000-gallons from 221-T.
ARH-1666 D	1970	Oct. to Dec.	534	293	539	233	370	32	T-112 Transferred 119,000-gallons to REDOX Evaporator. Received 138,000-gallons from 221-T.
ARH-2074 A	1971	Jan. to March	534	293	540	233	395	32	T-112 Transferred 76,000-gallons to REDOX Evaporator. Received 100,000-gallons from 221-T.
ARH-2074 B	1971	April to June	534	293	545	233	298	32	T-112 Transferred 244,000-gallons to REDOX Evaporator. Received 147,000-gallons from 221-T.
ARH-2074 C	1971	July to Sept.	534	293	539	233	342	32	T-112 Transferred 116,000-gallons to REDOX Evaporator. Received 160,000-gallons from 221-T.
ARH-2074 D	1971	Oct. to Dec.	534	293	541	233	399	32	T-112 Transferred 87,000-gallons to REDOX Evaporator. Received 144,000-gallons from 221-T.
ARH-2456 A	1972	Jan. to March	535	293	538	233	378	32	T-112 Transferred 260,000-gallons to REDOX Evaporator. Received 237,000-gallons from 221-T.
ARH-2456 B	1972	April to June	535	293	540	233	404	32	T-112 Transferred 146,000-gallons to REDOX Evaporator. Received 174,000-gallons from 221-T.

Reference	Year	Period	T-110	T-110	T-111	T-111	T-112	T-112	Comments from Reference Document
			Waste Volume (x 1000 gallons)						
			Total	Sludge	Total	Sludge	Total	Sludge	
PPD-493-7-DFL									REDOX Evaporators shutdown 7-5-1972. Waste to be stored in tanks and processed in 242-T or 242-S evaporators.
ARH-2456 C	1972	July to Sept.	535	293	539	233	468	32	T-112 received 136,000-gallons of decontamination waste from 221-T Plant. T-112 transferred 71,000-gallons to U-107.
ARH-2456 D	1972	Oct. to Dec.	535	293	539	233	329	32	T-112 received 148,000-gallons of decontamination waste from 221-T Plant. T-112 transferred 286,000-gallons to U-107.
ARH-2794A	1973	Jan. to March	536	293	536	233	217	32	T-112 Transferred 332,000-gallons to 107-U. T-112 received 199,000-gallons from 221-T and 20,000-gallons from 301-T catch tank
ARH-2794 B	1973	April to June	536	293	536	233	508	32	T-112 Transferred 180,000-gallons to 107-U. T-112 Received 120,000-gallons from 221-T and 350,000-gallons from 106-T.
ARH-2794 C	1973	July to Sept.	531	293	535	233	505	32	
ARH-2794 D	1973	Oct. to Dec.	531	293	534	233	505	32	221-T Decontamination solution routed to U-107
ARH-CD-133A	1974	Jan. to March	531	293	534	233	505	32	
ARH-CD-133B	1974	April to June	475	293	490	233	236	32	Transferred 59,000-gallons from T-110 to S-110, 28,000-gallons from T-111 to S-110, 14,000 from T-111 to T-109, and 273,000 from T-112 to S-110.
ARH-CD-133C	1974	July to Sept.	483	293	485	485	101	32	Transferred 8,000-gallons of water into T-110. Transferred 136,000-gallons from T-112 to S-110.
ARH-CD-133D	1974	Oct. to Dec.	483	466	488	488	101	32	T-111 removed from service.
ARH-CD-336A	1975	Jan. to March	483	466	488	488	101	32	
ARH-CD-336B	1975	April to June	483	466	488	488	101	32	
ARH-CD-336C	1975	July to Sept.	483	466	488	488	101	32	
ARH-CD-336D	1975	Oct. to Dec.	483	466	488	488	101	32	

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ORIGIN OF WASTES IN THE B-200 AND T-200 SERIES SINGLE-SHELL TANKS

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Abstract: Tanks B-201 through B-204 and T-201 through T-204 received waste from the plutonium concentration activities conducted in the 224-B and 224-T buildings from October 1946 through June 1952. Tanks B-201 through B-204 also received miscellaneous flush solutions from deactivation activities conducted in the 221-B Bismuth Phosphate Plant and the 224-B building.

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Tank Farm Contractor (TFC) RECORD OF REVISION	(1) Document Number RPP-13300	Page 1

(2) Title
Origin of Wastes in the B-200 and T-200 Series Single-Shell Tanks

Change Control Record

(3) Revision	(4) Description of Change - Replace, Add, and Delete Pages	Authorized for Release (5) Resp. Engr. (print/sign/date)	(6) Resp. Mrg. (print/sign/date)
Rg 1	Add - Section 2.1: Added discussion that plutonium precipitate separated from uranium and fission products was washed three times and the wash water combined with the uranium and fission product solution. Add - Section 2.1: Spent nuclear fuel reprocessing completed in the 221 BiPO4 process when plutonium was separated from the metal waste. Add: Figure 1: Added "Wastes Streams from" Bismuth Phosphate Process to figure title. Add: Section 2.2.1: Included the word "Plant" after "221" designation. Included the word "building" after "224" designation. Add - Section 3: Expanded discussion on types of records reviewed and information available in each of these records. Add - Section 3.2: Discussed the dissolvers metal waste equipment and other equipment in the 221-B Plant were flushed with nitric acid from July 1952 through September 1952, which would have removed any residual high-levle waset present in this equipment. Therefore, equipment flushing conducted after the nitric acid flushing would not have generated high-level waste. Discussed the water flushes of equipment conducted during 1954 through 1955 were discharged to cribs and cited reference documents. Add - Sections 3.4.1.1 & 3.4.2: Discussed certain historical records were not declassified until the 1990's which would have prevented the authors of WHC-MR-0132 and LA-UR-97-311 from knowing that the B-200 and T-200 series tanks were actually placed in service in 1946 instead of 1952. Add - Section 4.0: Discussed Hanford Site best basis inventories (BBIs) for tank	Michael E Johnson 10/18/04 Michael E. Johnson	S Mackey 12/15/04. STEWART MACKAY

GMP 12/15/04

Tank Farm Contractor (TFC) **RECORD OF REVISION (continued)**	(1) Document Number RPP-13300	Page 2

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	wastes. Indicated analyses of core samples and waste templates are the basis for the gross alpha and transuranic elements concentrations reported. Replaced the B-201 and B-202 inventories for transuranic elements with the B-200 and T-200 series tanks BBIs for transuranic elements.		

RPP-13300
Revision 1

ORIGIN OF WASTES IN THE B-200 AND T-200 SERIES SINGLE-SHELL TANKS

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SUMMARY

A review of historical documents was conducted to determine the origin of the wastes stored in single-shell tanks 241-B-201 through 241-B-204 and 241-T-201 through 241-T-204. This review was conducted to support disposition of the wastes in these tanks.

The wastes stored in tanks 241-B-201 through 241-B-204 and tanks 241-T-201 through 241-T-204 were determined to originate from plutonium concentration activities conducted from October 1946 through June 1952 in the 224-B and 224-T Concentration buildings. The 224-B and 224-T Concentration buildings received the plutonium nitrate solution that was separated from the irradiated reactor fuel as part of reprocessing activities conducted in the 221-B and 221-T Bismuth Phosphate plants. Tanks 241-B-201 through 241-B-204 also received miscellaneous flush solutions from deactivation activities conducted at the 221-B Bismuth Phosphate Plant and 224-B Concentration building. No other types of waste were transferred to these tanks.

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LIST OF TERMS

1C	first cycle of the decontamination process
2C	second decontamination cycle wastes
Ci	Curies
CW	coating waste
kg	kilograms
ηCi/g	nanocuries per gram
MW	metal waste
NCRW	neutralized cladding waste
PAS	PUREX acidified sludge
PSN	PUREX supernate neutralized
PSS	PUREX sludge supernate
PUREX	plutonium-uranium extraction
REDOX	reduction-oxidation
RSN	REDOX supernate neutralized
TBP	tri-butyl phosphate
TIC	total inorganic carbon
TOC	total organic carbon
WSTRS	Waste Status and Transaction Record Summary
μCi/g	microcuries per gram
μg/g	micrograms per gram

1.0 INTRODUCTION

This document discusses the origins of wastes presently stored in single-shell tanks 241-B-201 through 241-B-204 and tanks 241-T-201 through 241-T-204. Section 2.0 provides a description of the different types of wastes that were generated at the Hanford Site chemical processing plants and transferred to the underground storage tanks. A basic understanding of the different types of wastes that were generated at the Hanford Site is provided for the reader to comprehend the waste types transferred to tanks 241-B-201 through 241-B-204 and tanks 241-T-201 through 241-T-204, as discussed in Section 3.0. Section 4.0 summarizes the waste types that were transferred into these tanks.

2.0 TYPES OF TANK WASTE GENERATED AT THE HANFORD SITE CHEMICAL PROCESSING PLANTS

There are 149 single-shell tanks and 28 double-shell underground storage tanks located at the Hanford Site. These tanks received supernatants and precipitated sludges originating from the reprocessing of spent nuclear fuels, research and development, plutonium processing, and waste management activities.

There were numerous spent nuclear fuel reprocessing, research, and development, plutonium processing, and waste management activities conducted at the Hanford Site starting in 1944. 221-T Plant (T-Plant), first used for reprocessing of spent nuclear fuel in December 1944, operated until March 1956 using the bismuth phosphate process. The 221-B Plant (B-Plant) reprocessed spent nuclear fuel from April 1945 to June 1952 using the bismuth phosphate process. The bismuth phosphate process was based on carrier precipitation batch chemistry. The plutonium product solutions from the B-Plant and T-Plant were transferred to the 224-B and 224-T buildings for concentration. B-Plant was later renovated and used from 1963 through 1986 to recover the fission products cesium and strontium from the wastes stored in single-shell tanks.

Later, B- and T-Plants were replaced by the REDOX (reduction-oxidation) and PUREX (plutonium-uranium extraction) plants using continuous solvent extraction processes for separating uranium and plutonium from dissolved, spent nuclear fuels. The REDOX plant operated from January 1952 through November 1966 and PUREX operated intermittently from January 1956 to early 1990). Uranium was recovered from the wastes stored in the single-shell tanks from operation of the bismuth phosphate plants using a tributyl phosphate solvent extraction process in the tri-butyl phosphate (TBP) Plant (221-U building). The Hot Semiworks, building 201-C, was operated from 1949 through 1967 as a research and development facility for many of the Hanford Site chemical processes (e.g., REDOX, TBP, B-Plant strontium separations, PUREX process tests). All of these facilities generated numerous sources of radioactive mixed wastes that are stored in the single-shell tanks and double-shell tanks.

In addition to the operations conducted in the processing plants, there were numerous activities conducted within the underground storage tanks, including evaporation, cesium precipitation using ferrocyanide, and discharge of supernatants to underground cribs. These spent nuclear fuel reprocessing, research and development, plutonium processing, and waste management activities resulted in the mixing and alteration of the different waste types within several (but not all) of the 149 single-shell tanks and 28 double-shell tanks.

The spent nuclear fuel reprocessing, research and development, plutonium processing, and waste management activities conducted in the processing plants are described in the following sections. Refer to DOE/RL-97-02; *National Register of Historic Places Multiple Property Document Form - Historic, Archaeological and Traditional Cultural Properties of the Hanford Site, Washington February* for additional details on these processes.

As will be shown in Section 3.0, none of the spent nuclear fuel processing wastes from these operations were received into tanks 241-B-201 through 241-B-204 or 241-T-201 through 241-T-204. These tanks only received waste from the 224-B and 224-T plutonium concentration buildings. The B-200 series tanks also receive wastes from equipment cleaning. The process operations conducted in the 221-B, 224-B, 221-T and 224-T are discussed in the following subsections to provide an understanding of the waste types generated in these facilities.

2.1 BISMUTH PHOSPHATE PROCESS: B- AND T-PLANTS

B- and T-Plants were constructed in 1944 through 1945 to separate plutonium from spent nuclear fuel using the bismuth phosphate process. Figure 1 shows a summary of the 221-B and 221-T Plant bismuth phosphate process, which is referred to throughout this discussion.

In the bismuth phosphate process, the aluminum cladding of spent nuclear fuel elements was dissolved in boiling sodium nitrate solution, to which sodium hydroxide was slowly added (HW-10475-C, page 403). The cladding removal waste, sometimes referred to as Coating Waste (CW), was transferred to single-shell underground storage tanks. (See item [1] in Figure 1)

Reprocessing of the spent nuclear fuel commenced with the dissolution of the uranium fuel. The fuel element uranium cores (see item [2] in Figure 1) were dissolved in nitric acid (HW-10475-C, chapter IV, page 405). Water and sulfuric acid were added to the dissolved uranium metal solution and the mixture was then transferred to the plutonium extraction section. The sulfuric acid formed a uranyl sulfate complex that prevented uranium precipitation as a phosphate in the subsequent plutonium extraction step (HW-10475-C, page 418).

Plutonium was extracted from the acid solution by addition of bismuth nitrate and phosphoric acid to form a bismuth phosphate carrier precipitate (HW-10475-C, page 503). The plutonium and bismuth phosphate carrier precipitate was centrifuged and washed three times with water to separate the acidic supernatant from the plutonium precipitate, (see item [3] in Figure 1). The acidic solution remaining after the plutonium precipitation contained about 99 percent of the uranium, about 90% of the fission products. This separation process also removed and reduced the gamma radiation activity level in the plutonium precipitate by a factor of 10. However,

zirconium is phosphate insoluble and zirconium-95 (10 percent of the activity) stayed with the plutonium product. The acidic uranium solution was then neutralized and transferred to the underground single-shell tanks as Metal Waste (MW). Recent laboratory testing of the bismuth phosphate flowsheet confirms this partitioning of radionuclides (internal letter 7G300-02-NWK-024, "Bismuth Phosphate Process Radionuclide Partition Factors for the Hanford Defined Waste Model"). The laboratory tests indicate the percentage of cecium-137 and strontium-90 partitioned to the metal waste may have been as high as 100 percent and 89 percent respectively.

After separating and washing the plutonium precipitate from the metal waste, reprocessing of spent nuclear fuel was completed in the 221 Plant Bismuth Phosphate process. Plutonium decontamination was conducted in the remainder of the 221 Plant Bismuth Phosphate process. The plutonium bearing cake was dissolved in nitric acid and further decontamination of the plutonium to separate fission products was conducted (HW-10475-C, chapter VI). Sodium bismuthate, sodium dichromate, or potassium permanganate was added to oxidize the plutonium to the +6 valence-state. This step caused the bismuth phosphate to precipitate phosphate insoluble fission products ("by-product precipitation"), leaving the plutonium in solution. The precipitate was separated from the plutonium-bearing solution using centrifuges and washed to remove soluble plutonium. The plutonium was reduced to the +4 valence state to form a precipitate that could be separated from the remaining soluble fission products by centrifugation.

The fission products separated from the plutonium product during this first cycle of the decontamination process (designated as 1C) were combined with the coating removal waste and transferred to single-shell tanks. The 1C waste (see item [4] in Figure 1), contained approximately 10 percent of all fission products and approximately 1.4 percent of the plutonium present in the original fuel charged to the plant (HW-23043, pages 20 and 22). After 1951, the Bismuth Phosphate process flowsheet was modified to include cerium and zirconium scavenger precipitation in the 1C by-product step to remove lanthanide and zirconium radionuclides from the plutonium product (HW-23043, page 16).

The plutonium solids from the first decontamination cycle were dissolved in nitric acid. A second decontamination cycle (see item [5] in Figure 1) was conducted to reduced the gamma activity level by a factor of 10,000 from that in the previous dissolved metal solution, giving an overall process decontamination factor of 100,000 below that of the original solution (HW-10475-C, page 627). The second decontamination step essentially repeated the steps previously described for the first cycle decontamination. The second decontamination cycle wastes (designated as 2C) were also transferred to the single-shell tanks. The 2C waste contained less than 0.1 percent of the uranium and fission products and about 0.4 percent of the plutonium present in the original fuel charged to the plant (HW-23043, pages 26 and 28). The plutonium product from the bismuth phosphate process was subsequently transferred to the 224-B or 224-T building for concentration.

Table 1 provides the estimated compositions of the neutralized CW, MW, 1C, and 2C waste solutions generated from the bismuth phosphate plants based on the October 1, 1951 flowsheet (HW-23043). Additional analyses of the supernatant fraction of MW, 1C, and 2C that was stored in single-shell tanks are provided in Tables 2 and 3. These sample analyses support previous statements regarding the partitioning of fission products to the various Bismuth Phosphate Plant

Waste Streams. Specifically, 90% of the fission products were partitioned to the metal waste as evident by the Cs-137 concentration provided in Table 3. About 10% of the fission products partitioned to the 1C waste, as demonstrated by the gross beta and gross gamma radionuclides analyses provided in Table 2 and the Cs-137 analyses provided in Table 2. The 2C waste contained less than 0.1% of the fission products, as evident from information provided in Table 2.

Figure 1 Waste Streams from Bismuth Phosphate Process

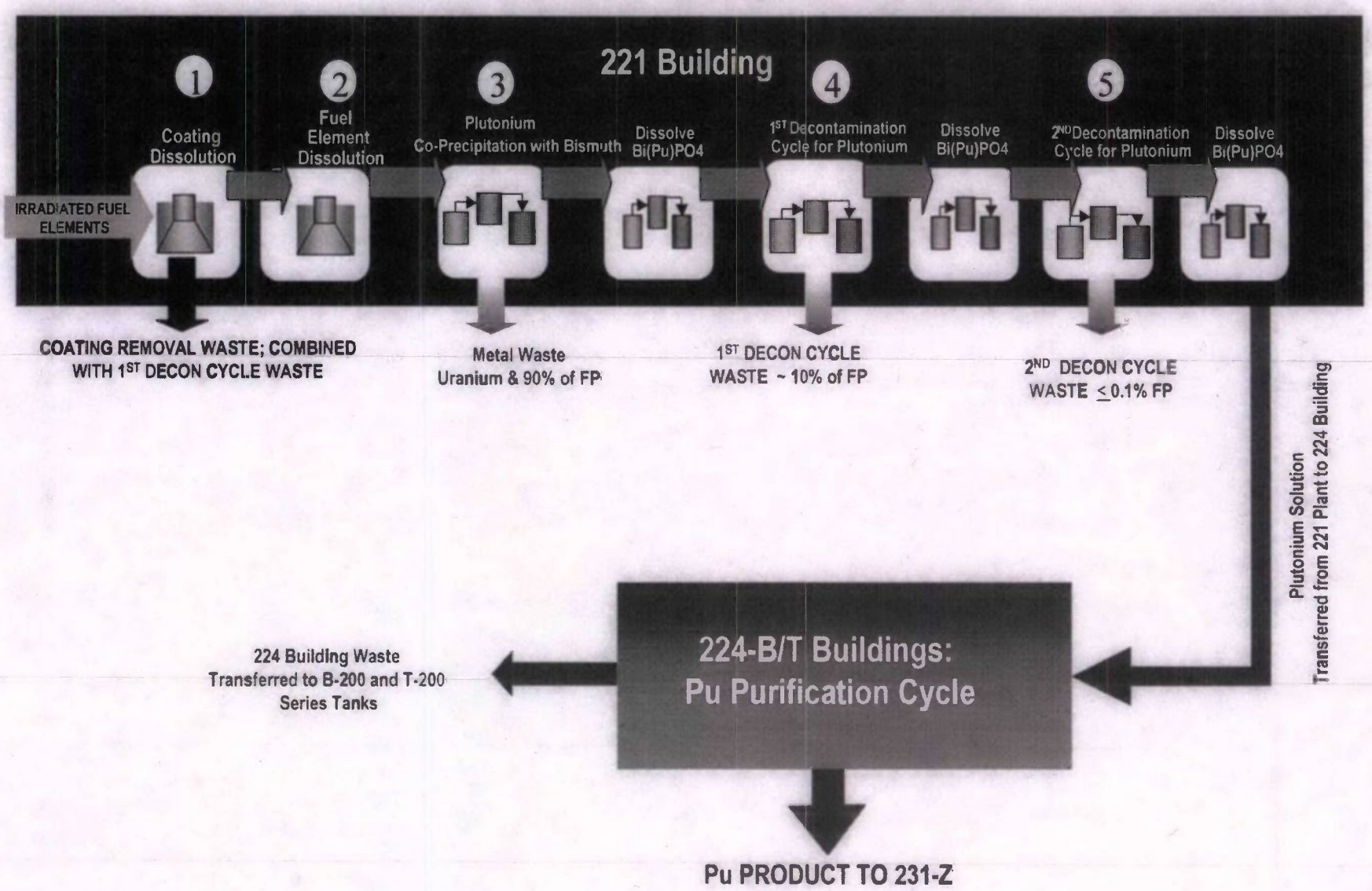


Table 1 Estimated Composition of Bismuth Phosphate Plant Wastes
From October 1, 1951 Flowsheet [1]

Analyte[2]	Coating Removal Waste[7]	Metal Waste[7]	First Decontamination Cycle (1C) Waste	Second Decontamination Cycle (2C) Waste	224 Building Waste
Plutonium	3.3E-04	2.0E-04	6.0E-07 [4]	1.6E-07 [5]	1.68E-04[6]
Uranium	0.15		0.235 [4]	Not reported	2.04E-05
Gamma	6.6E+04	1.3E+07	2.3E+06 [4]	1.13E+04 [5]	1.13E+02[6]
Sodium Aluminate ($NaAlO_2$)	95.1				
Sodium Hydroxide (NaOH)	43.6				
Sodium Nitrate ($NaNO_3$)	61.8				
Sodium Nitrite ($NaNO_2$)	56.0				
Sodium Silicate ($NaSiO_3$)	4.3				
Uranyl nitrate (UHN) [3]		132			
Fluorine (F)					5.6
Nitrate (NO_3)		9.7	93.1	61.3	42.4
Sulfate (SO_4)		24.4	4.73	3.61	0.35
Phosphate (PO_4)		25.2	26.2	23.0	3.05
Sodium (Na)		83.2	47.3	36.7	36.8
Bismuth (Bi)			2.59	1.31	1.18
Cerium (Ce)			0.030		
Lanthanum (La)					0.49
Manganese (Mn)					0.33
Zirconium (Zr)			0.030		
Iron (Fe)			1.37	1.82	
Chrome (Cr)			0.16	0.06	0.17
Ammonia (NH_4)			1.98	1.71	0.12
Silicon Hexa-Fluoride (SiF_6)			4.35	3.67	
Volume per Batch (gallons)	795	2,380	2,040	2,090	2,200

Notes:

(1) See HW-23043

(2) Analyses are reported in grams per liter, except for gamma activity, which is counts/minute/mL.

(3) HW-23043 page 31 notes that uranium is not actually present in this form, but is probably as $NaUO_2PO_4$ and $Na_4(UO_2)_2CO_3$.

(4) Pu and Gamma concentrations were calculated from the compositions of tanks 13-4 and 14-3 (HW-23043 pages 20 and 22).

(5) Pu and Gamma concentrations were calculated from the compositions of tanks 18-4 and 19-3 (HW-23043 pages 26 and 28).

(6) Pu and Gamma concentrations were calculated from the compositions of tanks A-4, D-4, B-3, and F-8 (HW-23043 pages 39, 44, 48, and 54)

(7) The coating waste batch size is based on 6,600-lbs uranium, but that the metal waste dissolution batch size is based on 2,200-lbs uranium.

Table 2 Analyses of Bismuth Phosphate Process Supernatants

Waste Type[1,2]	Tank	pH	Pu µGm/liter	Gross Beta millicuries/liter	Gross Gamma millicuries/liter	Date Sampled
Metal Waste	T-101	10.1	70	200[5]	70[5]	12-12-1946
Metal Waste	T-101	10	35	110	25	7-01-1947
Metal Waste	T-102	9.9	60	120	20	7-01-1947
Metal Waste	T-103	9.8	60	150	20	7-01-1947
1C/CW	B-109	9.9	40	0.65	0.28	3-18-1947
1C/CW	C-112	9.9	12	12	4.4	3-18-1947
2C	B-111	6.9	7.2E-02	2.0E-03	3.0E-03	7-1-1947
2C	B-112	6.8	4.32E?? [3]	1.5E-03	3.0E-03	7-1-1947
Waste Type	Tank	pH	Pu µGm/liter	Gross Beta Counts/minute/cc	Gross Gamma Counts/minute/cc	Date Sampled
2C	T-110	Not reported [4]	15	4.9E+04	30	7-13-1945
2C	T-110	9.8[4]	19	6.9E+04	55	7-25-1945
2C	B-110	9.6[4]	8.5	7.0E+04	55	7-25-1945

Notes:

[1] See HW-10728 and HW-3-3220.

[2] Solids formed in each of wastes, settling to the bottom of each tanks. These sample analyses are for the supernatant only and are not representative of the sludges.

[3] The reported Pu sample analyses for tank B-112 seems to be in error and lacking an exponent in HW-10728.

[4] Prior to October 1945, the 2C waste was neutralized to a pH of approximately 10. The waste collected in tanks 241-T-110, 241-T-111, and 241-T-112 were neutralized to about pH 7 after October 1945 to precipitate bismuth and plutonium (HW-3-3220, page 13).

[5] Reduction in the gross gamma and beta analyses for the metal waste in tank T-101 from sampling in 12-12-1946 to 07-01-1947 is due to decay of short-lived fission products.

Table 3 Radionuclide Analyses of Metal Waste and First Decontamination Cycle Waste

Tank	Date Filled	Pu µgm/cc	Gross Beta µCi/cc	Gross Gamma µCi/cc	Sr µCi/cc	Cs µCi/cc	Ru µCi/cc	Rare Earths + Y – Ce µCi/cc	Ce µCi/cc	Nb µCi/cc	Zr µCi/cc	Te µCi/cc
Analyses of Metal Waste Supernatant Following Uranium Extraction [1]												
C-106	Not specified				0.44	54.2						
BX-108	Not specified				0.26	132.4						
BX-109	Not specified				1.08	56.3						
C-112	Not specified				1.20	25.8						
C-109	Not specified				0.46	40.7						
C-111	Not specified				0.10	34.5						
Average Concentrations for Metal Waste					**0.59**	**57.3**						
Analyses of First Decontamination Cycle Waste Mixed with Coating Removal Waste Supernatant [2]												
B-107	8-1945	1.7E-02	0.135	0.055	0.011	0.10						
T-107	9-1945	1.5E-03	0.170	0.093	0.0013	0.20						
B-108	12-1945	2.0E-02	0.183	0.044	0.022	0.12						
T-108 (Top)	12-1945	2.0E-02	0.25	0.073	0.012	0.17	0.0066	0.047	0.007	0.0018	0	1.2E-05
T-108 (Bottom)	12-1945	2.0E-02	0.25	0.070	0.012		0.0065	0.029	0.0066	0.0024	0	3E-05
T-109	3-1946	2.6E-03	0.14	0.082	0.00038	0.15						
B-109	4-1946	1.8E-02	0.16	0.051	0.01	0.11						
T-104 (Top)	7-1946	3E-03	0.51	0.130	0.00013	0.13	0.058	0.004	0.051	0.028	0.010	2.4E-05
T-104 (Bottom)	7-1946	3E-03	0.52	0.160	0.00037		0.059	0.003	0.050	0.028	0.015	3.6E-05
C-110	8-1946	2E-03	0.14	0.0067	0.00026	0.11						
C-111	11-1946	4.2E-03	0.16	0.069	0.01	0.13						
C-112	4-1947	3.1E-03	0.14	0.064	0.005	0.13						
U-110	4-1947	2.1E-04	0.13	0.069	0.00011	0.17						
U-111	10-1947	3.4E-04	0.12	0.060	0.00023	0.14						
TX-109 [3]	9-1949	2.7E-05	2.8	2.2	0.00087	0.27	0.34	0.0085	0.0035	0.34	1.2	8E-05
Average Concentrations for 1C / CW		**7.67E-03**	**0.39**	**0.22**	**0.0058**	**0.15**						

Notes:

[1] HW-36717, Decontamination of Uranium Recovery Process Stored Wastes Interim Report, May 16, 1955, W. W. Schulz, General Electric Company, Richland Washington.

[2] HW-20195, Radioactive Content of Stored Bismuth Phosphate First Cycle Waste Supernatants, February 5, 1951, General Electric Company, Richland Washington.

[3] Tank TX-109 exhibits higher gross beta and gross gamma radioactivity since this tank was sampled shortly after filling and the short-lived fission products (e.g., Ru, Nb, and Zr) had not decayed appreciably.

2.1.1 224-B and 224-T Concentration Buildings

The process steps executed in the 224 Concentration buildings were as follows (HW-10475-C, chapter VII and HW-23043, pages 34 to 55):

- The starting batch size received from the 221 Plant was 330 gallons.
- Plutonium solution from the 221 Plant was oxidized with sodium bismuthate to convert the plutonium to the +6 valence state.
- Phosphoric acid was added to produce a bismuth phosphate ($BiPO_4$) precipitate, with the plutonium still in solution. At this point, operators wanted to get rid of all the $BiPO_4$.
- The solution and precipitate were separated by centrifugation.
- Nitric acid was added to dissolve the $BiPO_4$ precipitate, with this solution removed as waste.
- Potassium permanganate ($KMnO_4$) was added to the plutonium solution to ensure all the plutonium was in the +6 valence state.
- Hydrogen fluoride and lanthanum salts were added to the plutonium solution producing a lanthanum fluoride precipitate. Fission products were carried with the lanthanum. This precipitate contained all the lanthanides (cerium, lanthanum, etc.) and residual ruthenium, samarium, europium, americium, and curium that the $BiPO_4$ could not carry out of the stream.
- The lanthanum fluoride precipitate was dissolved in nitric acid, neutralized with sodium hydroxide, and sent to waste storage tanks.
- Oxalic acid was added to the plutonium solution collected from the lanthanum fluoride precipitation step to reduce the plutonium to the +4 valence-state.
- Hydrogen fluoride and lanthanum salts were added to the plutonium solution producing a lanthanum fluoride and plutonium fluoride precipitate. The precipitate was centrifuged to collect the solids.
- Potassium hydroxide was added to convert the plutonium fluoride / lanthanum fluoride precipitate into lanthanum hydroxide and plutonium hydroxide solids.
- After centrifuging to separate the lanthanum hydroxide and plutonium hydroxide solids, these solids are reacted with nitric acid solution to dissolve the lanthanum and plutonium. The plutonium nitrate / lanthanum nitrate solution product was now ready for transfer to the 231-Z building or 234-5 building.

By this time, each original 330-gallon batch of plutonium-bearing solution that had entered the 224 buildings was concentrated down to eight gallons. The liquid waste (designated as "224") from the lanthanum fluoride and barium sulfate precipitation process was neutralized and transferred to the single-shell tanks. Table 1 provides the estimated compositions of the neutralized 224 waste solutions based on the October 1, 1951 flowsheet (HW-23043).

3.0 ORIGINS OF WASTE IN TANKS 241-B-201 THROUGH 241-B-204 AND 241-T-201 THROUGH 241-T-204

This section provides a brief description of tanks 241-B-201 through 241-B-204 and 241-T-201 through 241-T-204, and a summary of the documented waste transfers into these tanks. In order to determine the origins of the wastes presently stored in these tanks, declassified historical reports for the Hanford Site were reviewed. Documents reviewed included the Hanford site contractors' monthly reports (1945 through 1975), Army Corp of Engineers monthly reports (December 1944 through December 1946), U. S. Atomic Energy Commission monthly reports (1947 through 1954), waste disposal reports (1948 through 1975), tank farm waste status summary reports, and miscellaneous letters and technical reports. While no records were located that identify individual transfers of waste, the above cited reports do provide a compendium of information that supports the discussion of historical waste types transferred into the B-200 and T-200 series tanks.

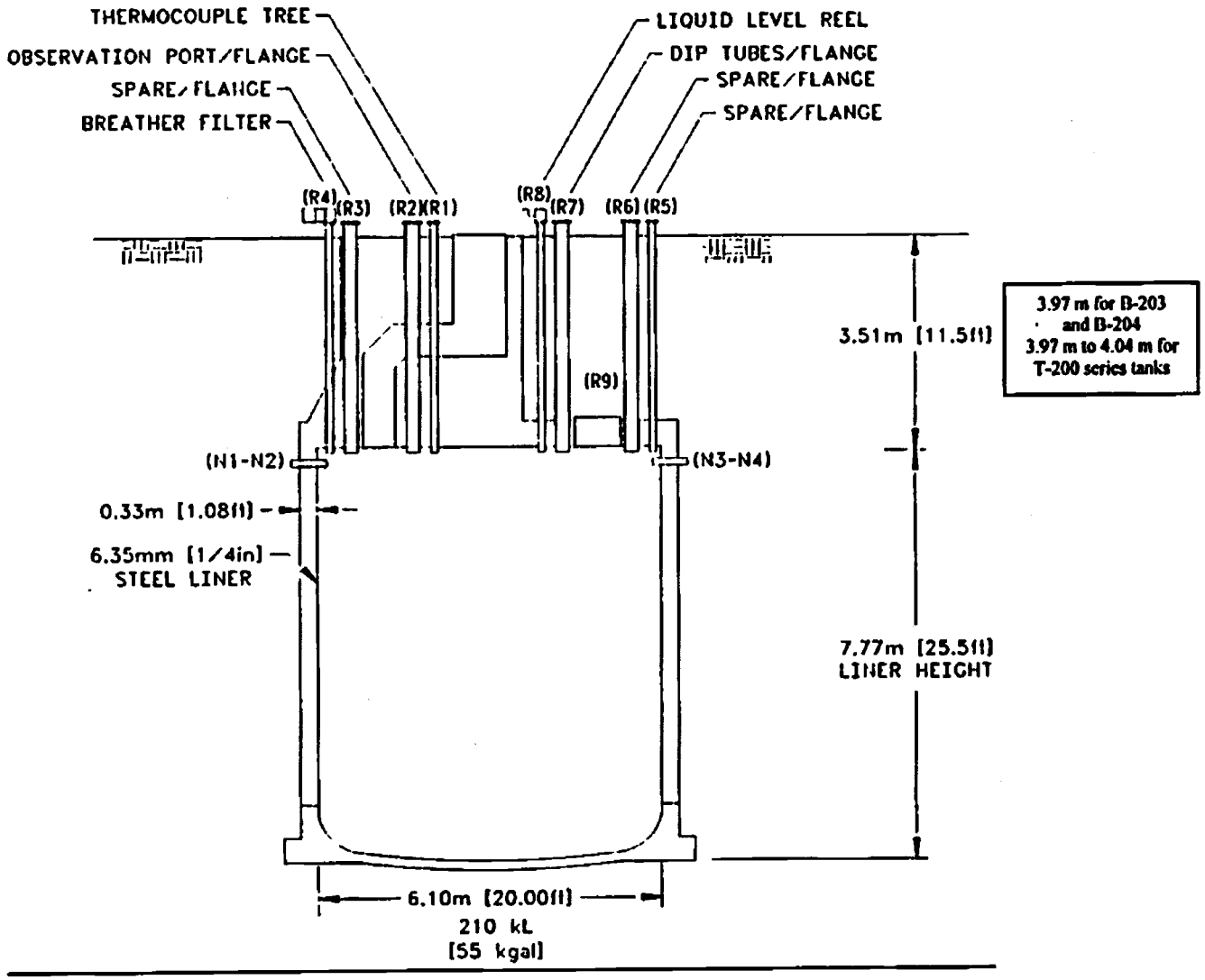
The Hanford site contractors' monthly reports for January 1945 through July 1951 list the volume of waste stored in the single-shell tanks, with the exception of the B-200 and T-200 series single-shell tanks. No records were located that provided the volume of wastes stored in the single-shell tanks from August 1951 through February 1952. Beginning in March 1952, waste transfers and the volume of waste stored in each single-shell tank were reported for each tank in a waste status summary report. Evidence of the waste types transferred to the B-200 and T-200 series single-shell tanks is provided in the Hanford site contractors' monthly reports, waste disposal reports, and miscellaneous letters and technical reports cited in the following sections.

With the exception of the waste status summary reports, all reports cited in this section are available electronically from the Hanford Declassified Document Retrieval System at http://www2.hanford.gov/declass/. Full-text copies of the waste status summary reports cited in this section are provided in Appendix A. The results of the present review of historical records are compared with a previous review of historical report that was conducted in 1980 as part of WHC-MR-0132, *A History of the 200 Area Tank Farms*.

3.1 DESCRIPTION OF TANKS 241-B-201 THROUGH 241-B-204 AND 241-T-201 THROUGH 241-T-204

Single-shell tanks 241-B-201 through 241-B-204 (B-200 series) and tanks 241-T-201 through 2410T-204 (T-200 series) were originally constructed in 1944 as part of the Manhattan Project (HW-10475-C, chapter IX). The B-200 and T-200 series tanks are twenty-foot diameter underground tanks made of reinforced concrete with a steel liner on the bottom and sides, as depicted in Figure 2. Each tank has a design capacity of 55,000 gallons at a liquid depth of twenty-four feet. The 200 series tanks are grouped together with twelve larger capacity single-shell tanks (100 series) to comprise a tank farm.

Figure 2 Cross Section of 200 Series Single-Shell Tank.



3.2 WASTE TRANSFERS INTO TANKS 241-B-201 THROUGH 241-B-204

This section discusses the date and source of wastes that were transferred into tanks 241-B-201 through 241-B-204. Tanks 241-B-201 through 241-B-204 did not receive any high-level wastes. These tanks received transuranic waste from operations conducted at the 224-B plutonium concentration building and equipment decontamination waste. According to the *Hanford Technical Manual Section C* for the bismuth phosphate process (HW-10475-C, pages 909 - 911), the metal waste solution from the bismuth phosphate processing plant (B-Plant) was originally planned to be decontaminated to separate fission products from the uranium using scavenger precipitation processes. This decontamination process was to be conducted in B-Plant with the precipitates being transferred to the 241-B-201 through 241-B-204 tanks. However, the metal waste decontamination process was never implemented and metal waste solution was *not* transferred into these tanks. Tanks 241-B-201 through 241-B-204 were unused until October 1, 1946.

Beginning in October 1946, tanks 241-B-201 through 241-B-204 were used as settling tanks for the solids that were contained in the 224-B Concentration building waste, with the liquid waste discharged to the 241-B-1 and 241-B-2 cribs. Prior to October 1946, the waste from the 224-B Concentration building was transferred to the 361-B settling tank and the liquid portion discharged to the 241-B-361, reverse-well. By September 1946, solids had accumulated in the 361-B settling tank to a point where the tank had reached its storage capacity, causing shutdown of 221-B and 224-B building operations, as reported in the Army Corp of Engineers monthly report for September 1946 (HAN-45800, page 77). A project was initiated in August 1946 to divert the 224-B Concentration building waste to tank 241-B-201 (HW-7-4640). The Army Corp of Engineers monthly report for October 1946 reports this project was completed on October 1, 1946, at which time a connection was made from the 224-B building waste transfer line to tank 241-B-201 (HAN-45800, page 87). A similar connection was also completed on October 14, 1946 from the 224-T Concentration building waste transfer line to tank 241-T-201. The *Hanford Engineering Works Monthly Report for October 1946* (HW-7-5362-DEL, pages 27 and 28) confirms that the 224-B building waste was routed to tank 241-B-201 in October 1946.

Tank 241-B-201 received waste from the 224-B Concentration building from October 2, 1946 through October 1948, after which the tank was considered filled with solids and the 224-B Concentration building waste was diverted to tank 241-B-204 (HW-11499, page 34). Tank 241-B-204 was connected in a cascade with tanks 241-B-203 and 241-B-202 (HW-10714-DEL, page 31). Liquid was gravity discharged from the last tank in the cascade, 241-B-202 to the 241-B-1 and 241-B-2 cribs. Solids contained in the 224-B building waste were allowed to settle in tanks 241-B-204 through 241-B-202. The cascade of tanks 241-B-204, 241-B-203, and 241-B-202 continued to receive 224-B Concentration building wastes until September 1952. The discharge of 224-B Concentration building waste from the B-202 tank to the 241-B-1 and 241-B-2 cribs is documented in HW-20583, *Process Waste Disposal Summary – 200 Areas September 1949 through December 1950*, HW-25301, *Process Waste Disposal Summary – 200 Areas January 1952 through June 1952*, HW-28121, *Release of Radioactive Wastes to Ground*, and HW-33591, *Summary of Liquid Radioactive Wastes Discharged to the Ground – 200 Areas July 1952 through June 1954.*

In addition to waste from the 224-B Concentration building, tanks 241-B-202 through 241-B-204 also received low-activity waste from tank 5-6 in 221-B Plant from October 3, 1947 through August 12, 1948 (HW-17088, page 31 and HW-38562, page 9). Tank 5-6 received low activity cell drainage as well as scrubber solution from the dissolvers in 221-B Plant (HW-10728, page 2). Tanks 241-B-201 through 241-B-204 were reported to have received a total of 22,300,000 liters of waste containing 2180 grams of plutonium and 4000 curies of fission products from May 1947 through January 1, 1950 (HW-17088, page 57). Approximately 7,400,000 liters of the waste transferred to tank 241-B-201 was low-activity waste from tank 5-6 in 221-B Plant (HW-17088, page 57). The low concentration of fission products in the 224 building and tank 5-6 waste, ~180 micro-curies per liter supports that no high-level waste was transferred along with the tank 5-6 waste to tanks 241-B-201 through 241-B-204.

In July 1952, B-Plant and the 224-B Concentration building were shut down because their processing capability was no longer needed and had been replaced by the 202-S REDOX facility. Beginning in July 1952, cleanout of B-Plant and the 224-B Concentration building was initiated, with the spent nuclear fuel dissolver heels removed from equipment in the 221-B building (HW-25227-DEL, pages Ed-1 and Ed-6). The dissolvers, metal waste equipment and other process equipment in the 221-B Plant were flushed with nitric acid solution from July 1952 through September 1952 to remove fission products and plutonium. The recovered plutonium solutions were processed through the normal bismuth phosphate flowsheet and wastes transferred to their normal disposal pathways (HW-25227-DEL page Ed-1 and Ed-6, HW-25533-DEL, pages Ed-1 and Ed-6, HW-25781-DEL, page Ed-1, and HW-26047-DEL, pages Ed-1 and Ed-5). Plutonium solutions derived from equipment cleanout activities in the 221-B Plant were processed in the 224-B Concentration building to recover the plutonium, with waste from the 224-B Concentration building transferred to the cascade of tanks 241-B-204, 241-B-203, and 241-B-202.

The waste solutions generated from nitric acid flushing of the 221-B Plant equipment were transferred to their normal disposal pathways. The B-200 series single-shell tanks only received waste from the 224-B Concentration building during the nitric acid flushing of the 221-B Plant equipment. Monthly reports for the 200 Area tank farms for April 1952 through September 1952 (HW-27838 and HW-27839) substantiate that only 224-B Concentration building waste was discharged to the cascade of tanks 241-B-204, 241-B-203, and 241-B-202 during the nitric acid flushing of 221-B Plant equipment.

High-level waste was removed from affected 221-B Plant equipment as a result of removing the heels from the dissolvers and nitric acid flushing of the equipment. Additional cleaning of the internal surfaces of piping and equipment in 221-B Plant and 224-B building was conducted using various chemical solutions and water, as described in HW-27774. This cleaning occurred from October 1952 through March 1953.

Flushes of metal waste, first decontamination cycle, and second decontamination cycle equipment were transferred to the cascade of tanks 241-B-110, 241-B-111, and 241-B-112, as documented in waste status summary reports for the 200 Area tanks farms for this period (HW-27840, HW-27841, HW-27842, and HW-27775). In November 1952, the cascade of tanks 241-B-204, 241-B-203, and 241-B-202 received flushes of 224-B building equipment and metal waste lines (HW-27840, page 20). In December 1952, they received flushes of 224-B building equipment and 221-B Plant section 9 metal waste tanks (HW-27840, page 28). In January, February and March 1953 they received flushes of 224-B building equipment and 221-B Plant sections 7 and 8 extraction equipment (HW-27841, page 9, HW-27842, page 9, HW-27775, page 9). As previously discussed, the dissolvers, metal waste equipment and other equipment in the 221-B Plant had been flushed with nitric acid from July through September 1952 and removed the high level waste from affected equipment. Therefore, the 221-B Plant equipment flushing conducted from October 1952 through March 1953 did not generated high-level waste.

Flushing of the B-Plant cells and wetting of process equipment with water was conducted in April 1953 through June 1, 1953 (HW-27932-DEL, page Ed-5; HW-28267-DEL, page Ed-5; and HW-28576-DEL, page Ed-5). These flush solutions were transferred to the cascade of tanks 241-B-110, 241-B-111, and 241-B-112, as documented in waste status summary reports for the 200 Area tanks farms for this period (HW-28043, HW-28377, and HW-28712). Additional decontamination of equipment in the 224-B Concentration building was also conducted in May through July 1953 with the flush solutions processed through T-Plant to recover plutonium (HW-28267-DEL, page Ed-5, HW-28576-DEL, page Ed-5, and HW-28906-DEL, page Ed-5).

In October 1954, approximately 50,000-gallons of water were transferred from the 221-B Plant through the cascade of tanks 241-B-201 through 241-B-204 (HW-33544, page 4 and HW-38562, page 9). In December 1954, tank 5-6 (low activity cell drainage in 221-B Plant) was reported as being routed to the cascade of tanks 241-B-204, 241-B-203, and 241-B-202, but no volume of waste was reported as being discharged to these tanks in the tank farm monthly waste status summary report (HW-34412, page 4). However, the June 1955 report for discharge of wastes to the ground (HW-38562, page 9) indicates approximately 750,000 liters (~198,000 gallons) of low-activity waste were discharged from 221-B Plant tank 5-6 from December 1954 through June 1955 to tanks 241-B-202 through 241-B-204 to the 241-B-1 and 241-B-2 cribs.

In July 1955, 224-B Concentration building flush water was reported as being routed to the cascade of tanks 241-B-204, 241-B-203, and 241-B-202, but no volume of waste was reported as being discharged to these tanks in the tank farm monthly waste status summary report (HW-38401, page 4). However, the June 1956 report for discharge of wastes to the ground (HW-44784, page 27) indicates approximately 653,000 liters (~172,500 gallons) of low-activity waste were discharged from 221-B Plant tank 5-6 from July 1955 through September 1955 to tanks 241-B-202 through 241-B-204 to the 241-B-1 and 241-B-2 cribs. Beginning in October 1955, the

low activity waste from 221-B Plant tank 5-6 was routed to the cascade of tanks 241-B-110, 241-B-111 and 241-B-112 and then to the 241-B-1 and 241-B-2 cribs (HW-44784, page 27).

During the period of 1954 through 1955, the 221-B Plant and 224-B Concentration building were being modified for restart as part of the so-called "4X Program" (HW-33903). The 4X Program was a program to operate all four separation facilities (221-B, 221-T, 202-S REDOX and 202-A PUREX Plants) simultaneously. The modifications conducted in the 221-B Plant and 224-B Concentration building resulted in the transfer of low-activity waste to tanks 241-B-202 through 241-B-204. However, the 4X Program was cancelled in March 1957 and the 221-B Separations Plant and 224-B Concentration building were placed in lay-away status (DDTS-Generated-491, "Lay-Away of the Bismuth Phosphate – TBP Plants and the Metal Waste Removal Facilities").

From 1957 through 1963, the 221-B Plant was converted for separating fission products of cesium-137 and strontium-90 from PUREX plant wastes. A flush of 7,500-gallons was transferred from the 221-B Plant to the cascade of tanks 241-B-204, 241-B-203, and 241-B-202 sometime January 1, 1962 through June 30, 1962 (HW-74647, page 4). This is the last transfer of any waste solutions into tanks 241-B-201 through 241-B-204. The 221-B Plant did not receive any waste for fission product separation until August 2, 1963 (HW-78817, page 5) and did not discharge fission product waste to tanks 241-B-201 through 241-B-204.

Tanks 241-B-201 through 241-B-204 did not receive any liquid wastes originating from the operation of the first cycle solvent extraction system, or equivalent, or the concentrated wastes from subsequent extraction cycles, or equivalent, in a spent fuel reprocessing facility. Rather, the wastes received into tanks 241-B-201 through 241-B-204 were from a *plutonium concentration* process (not a subsequent *extraction* process) performed in a *separate* facility and wastes from equipment decontamination activities conducted in the 221-B Bismuth Phosphate Plant and 224-B Concentration building.

3.3 WASTE TRANSFERS INTO TANKS 241-T-201 THROUGH 241-T-204

This section discusses the date and source of wastes that were transferred into tanks 241-T-201 through 241-T204. These tanks did not receive any high-level waste but did receive transuranic waste from operations conducted at the 224-T plutonium concentration building. According to the *Hanford Technical Manual Section C* for the bismuth phosphate process (HW-10475-C, chapter X, pages 909 - 911) the metal waste solution from T-Plant was originally planned to be decontaminated to separate fission products from the uranium using scavenger precipitation processes. This decontamination process was to be conducted in T-Plant with the precipitates being transferred to the 241-T-201 through 241-T-204 tanks. However, the metal waste decontamination process was never implemented and metal waste solution was *not* transferred into these tanks. Tanks 241-T-201 through 241-T-204 were unused until November 4, 1946.

Beginning on November 4, 1946, tank 241-T-201 was used as a settling tank for the solids that were contained in the 224-T Concentration building waste, with the liquid discharged to the 241-T-1 and 241-T-2 cribs (HW-33591, page 4). The waste from the 224-T Concentration building had been previously transferred to the 361-T settling tank and the liquid portion

discharged to the 241-T-361, reverse-well. By July 1946, solids had accumulated in the 361-T, settling tank to a point where the tank had reached its storage capacity (HAN-45800, page 67). A project was initiated in August 1946 to divert the 224-T Concentration building waste to tank 241-T-201 (HW-7-4640). The Army Corp of Engineers monthly report for October 1946 reports this project was completed on October 14, 1946, at which time a connection was made from the 224-T building waste transfer line to tank 241-T-201 (HAN-45800, page 87). The Hanford Engineering Works monthly report for October 1946 (HW-7-5362-DEL, page 27 to 28) confirms that a route was established for transfer of the 224-T building waste to tank 241-T-201 in October 1946, with waste transfer initiated on November 4, 1946.

Tank 241-T-201 received waste from the 224-T Concentration building from November 4, 1946 through May 24, 1949, after which the tank was considered filled with solids and the 224-T Concentration building waste was routed to tank 241-T-204 (HW-13561-DEL, page 41). The solids depth in tank 241-T-201 was reported as "twenty feet of rather compact sludge and approximately three feet of a light sludge" (HW-13561-DEL, page 41).

Tank 241-T-204 was connected in a cascade with tanks 241-T-203 and 241-T-202, (HW-10714-DEL, page 31). Liquid was gravity discharged from the last tank in the cascade, tank 241-T-202, to the 241-T-1 and 241-T-2 cribs. Solids contained in the 224-T Concentration building waste were allowed to settle in tanks 241-T-204 through 241-T-202.

The tanks 241-T-204, 241-T-203, and 241-T-202 cascade continued to receive 224-T Concentration building wastes until May 29, 1952, after which this waste was transferred to the cascade of single-shell tanks 241-T-110, 241-T-111, and 241-T-112 (HW-27838, page 17). Tanks 241-T-201 through 241-T-204 were considered filled with solids and taken out of service effect on May 29, 1952 (HW-27838, page 12).

Tanks 241-T-201 through 241-T-204 did not receive any liquid wastes originating from the operation of the first cycle solvent extraction system, or equivalent, or the concentrated wastes from subsequent extraction cycles, or equivalent, in a spent fuel reprocessing facility. Rather, the wastes are either from a *plutonium concentration* process (not a subsequent *extraction* process) performed in a *separate* facility (224-T Concentration building).

3.4 CURRENT REVIEW OF WASTE TRANSFER RECORDS COMPARED WITH OTHER REVIEWS

Historical records of waste transfers into, from and among the 200 Area tank farms were compiled and reported in WHC-MR-0132, LA-UR-96-3860, and LA-UR-97-311. Additional waste transfer records have been summarized for the B-200 series tanks in WHC-SD-WM-ER-310. These documents were reviewed and compared with the current analysis documented in this report to determine if significant discrepancies exist.

3.4.1 B-200 Series Tanks

These previous reviews to determine the tank contents for tanks 241-B-201 through 241-B-204 generally refer to WHC-MR-0132 and provide no new references regarding waste transfers into these tanks. Therefore, the current review of waste transfer records for tanks 241-B-201 through 241-B-204 was compared to that of WHC-MR-0132. Appendix B provides copies of the tabulated waste transfer records for tanks 241-B-201 through 241-B-204 from WHC-MR-0132.

In general, the waste transfer records summarized in WHC-MR-0132 are consistent with the information present in this document. However, there are three significant discrepancies that are discussed in the following sections.

3.4.1.1 Date Tanks Used to Receive Waste

The waste transfer records summarized in WHC-MR-0132 for tanks 241-B-201 through 241-B-204 do not indicate the presence of any waste in tanks 241-B-201 through 241-B-204 until the first quarter of 1952. This is inconsistent with historical documents discussed in this document, which indicates that these tanks received waste from the 224-B Concentration building beginning in October 1946. The Hanford site contractors' monthly reports for January 1945 through July 1951 list the volume of waste stored in the single-shell tanks, with the exception of the B-200 and T-200 series single-shell tanks. Evidence of the waste types transferred to the B-200 and T-200 series single-shell tanks is provided in the Hanford site contractors' monthly reports, waste disposal reports, and miscellaneous letters and technical reports cited in the following sections. These documents were classified until the early 1990's, which would have limited their availability. It is likely that these documents were unavailable to the authors of WHC-MR-0132 and LA-UR-97-311.

3.4.1.2 Cascade Operation of Tanks

Beginning in the first quarter of 1952, WHC-MR-0132 indicates that 224-B Concentration building waste (designated as "224") was transferred into tanks 241-B-201 through 241-B-204, which were operated as a cascade to a crib. However, this contradicts information published in HW-33591(*Summary of Liquid Radioactive Wastes Discharged to the Ground – 200 Areas July 1952 through June 1954*), which states that tank 241-B-201 received waste from the 224-B Concentration building only from October 2, 1946 through October 1948, after which the tank was considered filled with solids and the 224-B Concentration building waste was diverted to tank 241-B-204.

Tank 241-B-204 was connected in a cascade with tanks 241-B-203 and 241-B-202. Liquid was gravity discharged from the last tank in the cascade, tank 241-B-202, to the 241-B-1 and 241-B-2 cribs. Furthermore, tank farm waste status summary records for April through June 1952 (HW-27838) and July through September 1952 (HW-27839) indicate that only tanks 241-B-204

through 241-B-202 were active in a cascade with discharge to the crib. Again, this inconsistency in records does not affect the classification of the wastes in tanks 241-B-201 through 241-B-204.

3.4.1.3 Metal Waste Not Transferred to Tanks

WHC-MR-0132 identifies that metal waste (designated as "MW") was transferred into tanks 241-B-201 through 241-B-204 in the fourth quarter of 1952. WHC-MR-0132 indicates that metal waste was present in tanks 241-B-201 through 241-B-204 through the third quarter of 1953, after which time WHC-MR-0132 indicates that only "224" waste is present in these tanks. There is no explanation provided in WHC-MR-0132 to indicate that metal waste was removed from these tanks or that the earlier designation of metal waste being present in these tanks was incorrect.

It is highly unlikely that metal waste was transferred into tanks 241-B-201 through 241-B-204, since these tanks were active as a cascade that overflowed to the ground via a crib. Metal waste was not discharged to the ground because of the concentration of fission products and the economic value of the uranium. Because of the economic value of uranium, metal waste was kept segregated from other wastes. Document HW-33591 (*Summary of Liquid Radioactive Wastes Discharged to the Ground – 200 Areas July 1952 through June 1954*) does not indicate the disposal of any metal waste to these tanks or the ground (i.e. cribs).

Process records for waste transfers to the 200 Area tank farms indicate that metal waste generated from spent nuclear fuel reprocessing at 221-B Plant was transferred to tank 241-BY-112 in BY Tank Farm for the period of April 1952 through September 1952 (HW-27838, pages 9, 21, and 32 and HW-27839, pages 10, 21, 32). These same historical records indicate that only flushes of the metal waste lines and 221-B and 224-B building equipment were transferred to the cascade of tanks 241-B-204 through 241-B-202 during October 1952 through March 1953. Based on this information, WHC-MR-0132 incorrectly stated the presence of metal waste in tanks 241-B-201 through 241-B-204.

Further evidence that metal waste was not transferred into tanks 241-B-201 through 241-B-204 is provided by the analyzed composition of the sludges presently stored in these tanks, as reported in the Tank Waste Information Network (http://twins.pnl.gov/twins.htm). Key analytes present in the wastes stored in tanks 241-B-201 through 241-B-204 are summarized in Tables 4.

The inventory of key analytes (e.g., Pu-239, Cl, Fe, Na, NO_3, PO_4, SO_4, CO_3, and U) in the sludge phase of tanks 241-B-201 through 241-B-204 has been divided by the mass of the sludge in each tank and reported in the upper half of Table 4. The composition of metal waste that was contained in single-shell tanks 241-T-101, 241-U-101 and 241-U-102 is provided in the lower half of Table 4. It is evident from the analyses in Table 4 that the concentrations of Pu-239, Cl, and Fe in the sludges present in tanks 241-B-201 through 241-B-204 are one order of magnitude higher than the metal waste sludges. Also, the concentration of uranium in the sludges present in tanks 241-B-201 through 241-B-204 is three orders of magnitude lower than the metal waste sludges. This comparison clearly shows that metal waste sludge contained a higher uranium concentration and a lower plutonium concentration than the sludges present in tanks 241-B-201

to 241-B-204. The sludges present in tanks 241-B-201 to 241-B-204 do not have the characteristics of metal waste sludge. Furthermore, the concentrations of cesium-137 and strontium-90 sludge present in tanks 241-B-201 through 241-B204 are less than 2E-04 Ci/l and 4E-03 Ci/l, respectively. Metal waste or other high-level waste would have cesium-137 and strontium-90 concentration several orders of magnitude higher, as reported in Table 3.

3.4.2 T-200 Series Tanks

In general, the waste transfer records summarized in WHC-MR-0132 are consistent with the information present in this document. The waste transfer records summarized in WHC-MR-0132 (see Appendix B) and the Waste Status and Transaction Record Summary (LA-UR-97-311) report for tanks 241-T-201 through 241-T-204 do not indicate the presence of any waste in these tanks until the first quarter of 1952. This is inconsistent with historical documents discussed in the previous section, which indicate that these tanks received waste from the 224-T Concentration building beginning in November 1946. The Hanford site contractors' monthly reports for January 1945 through July 1951 list the volume of waste stored in the single-shell tanks, with the exception of the B-200 and T-200 series single-shell tanks. Evidence of the waste types transferred to the B-200 and T-200 series single-shell tanks is provided in the Hanford site contractors' monthly reports, waste disposal reports, and miscellaneous letters and technical reports cited in the following sections. These documents were classified until the early 1990's, which would have limited their availability. It is likely that these documents were unavailable to the authors of WHC-MR-0132 and LA-UR-97-311.

WHC-MR-0132 and LA-UR-97-311 both indicate that only 224-T building waste was received in these tanks and that these four tanks were removed from service in May 29, 1952. This is consistent with the current review of waste transfer records for these tanks.

Table 4. Comparison of Metal Waste and Tanks 241-B-201 to 241-B-204 Sludges.

	Pu-239	Cl	Fe	Na	NO_3	PO_4	SO_4	CO_3	U Total	Sample Event Description	Reference
	µCi/g	µg/g	µg/g	µg/g	µg/g	µg/g	µg/g	µg/g	µg/g		3
241-B-201	7.51E-01	1.68E+03	1.15E+04	3.42E+04	5.05E+04	1.45E+04	3.48E+02	1.61E+04	1.56E+02	Composite analysis of core sample	3
241-B-202	1.28E-01	8.64E+02	1.19E+04	3.93E+04	6.15E+04	1.12E+04	9.81E+03	9.81E+03	3.23E+02	Composite analysis of core sample	3
241-B-203	1.94E-01	8.39E+02	3.30E+03	2.89E+04	6.22E+04	6.65E+03	2.67E+02	3.34E+03	1.78E+02	Composite analysis of core sample	3
241-B-204	2.39E-01	6.46E+02	3.72E+03	2.63E+04	4.94E+04	7.06E+03	2.81E+02	6.73E+03	1.61E+02	Composite analysis of core sample	3
Metal Waste Sample Analyses											
U-101 (S1)	5.75E-02	5.00E+02	3.80E+02	1.18E+05		5.30E+04	1.10E+03	1.57E+05	2.88E+05	Near tank inlet, near top of sludge	1
U-101 (S2)	2.91E-02	7.00E+01	1.00E+02			3.30E+04	4.10E+03	2.08E+05	3.63E+05	Near tank outlet, 1-ft from bottom	1
T-101 (a)				1.16E+05	5.10E+04	1.11E+05	2.40E+04	3.00E+03	1.12E+05	6/1948, Near tank inlet, depth unknown	2
T-101 (b)				8.50E+04	1.60E+04	1.34E+05	2.20E+04	1.00E+03	1.34E+05	8/1948, Near tank inlet, depth unknown	2
T-101 (2)				1.04E+05		1.28E+05	1.90E+04	1.58E+04	1.62E+05	11/1948, Near tank inlet, 3-ft from bottom	2
T-101 (3)				1.16E+05		1.02E+05	1.20E+04	2.56E+04	2.04E+05	11/1948, Near tank inlet, 2-ft from bottom	2
T-101 (4)				1.37E+05		4.80E+04	8.00E+03	1.15E+05	3.63E+05	11/1948, Near tank inlet, 1-ft from bottom	2
T-101 (5)				1.51E+05		1.60E+04	8.00E+03	2.55E+05	3.46E+05	11/1948, Near tank outlet, 1-ft from bottom	2
T-101 (6)				1.46E+05		1.50E+04	6.00E+03	2.63E+05	3.65E+05	11/1948, Near tank outlet, just off bottom	2
U-101 (13)	2.70E-02			1.27E+05	2.00E+03	6.10E+04	2.80E+03	1.81E+05	3.29E+05	2/1949, Near tank inlet, 4-ft from bottom	2
U-101 (14)	1.29E-02			9.40E+04	3.00E+03	4.90E+04	2.05E+03	1.66E+05	2.63E+05	2/1949, Near tank inlet, 3-ft from bottom	2
U-101 (15)	2.11E-02			1.23E+05	3.50E+03	5.30E+04	2.54E+03	1.62E+05	2.73E+05	2/1949, Near tank inlet, 2-ft, 7-inches from bottom	2
U-101 (16)	2.59E-02			1.08E+05	2.00E+03	5.20E+04	3.03E+03	4.10E+04	2.66E+05	2/1949, Near tank outlet, 3-ft from bottom	2
U-102 (17)	1.83E-02			1.09E+05	3.00E+03	1.36E+05	5.08E+03	6.90E+04	1.06E+05	2/1949, Near tank inlet, 1-inch from bottom	2
U-102 (18)	6.61E-03			8.50E+04	2.00E+03	1.05E+05	9.26E+03	9.00E+04	1.52E+05	2/1949, Near tank outlet, 1-inch from bottom	2

Notes:

Ref. 1: HW-18492, *Settling and Dissolution Characteristics and Composition of Hanford Waste Metal Sludge.*
Ref. 2: Letter HW-14157, "Compilation of Data on Composition of Bismuth Phosphate Process Metal Wastes."
Ref. 3: Best Basis Inventory obtained from the Tank Waste Information System on October 7, 2002. Sechttp://twins.pnl.gov/twins.htm

4.0 TRANSURANIC ANALYSES OF WASTES

The Hanford Site prepares a Best Basis Inventory (BBI) estimate of the composition of the wastes stored in all 177 Hanford Site underground storage tanks. The BBI effort involves developing and maintaining waste tank inventories comprising 25 chemical and 46 radionuclide components in the 177 Hanford Site underground storage tanks. Waste sample analyses, process knowledge, and waste templates are used to create the BBIs. These BBIs provide waste composition data necessary as part of the River Protection Project (RPP) process flowsheet modeling work, safety analyses, risk assessments, and system design for retrieval, treatment, and disposal operations. Development and maintenance of the BBI is an on-going effort, with the current BBIs available electronically through TWINS, http://twins.pnl.gov/data/datamenu.htm.

The BBIs for the wastes contained in tanks 241-B-201 through 241-B-204 and 241-T-201 through 241-T-204 are based on analyses of core samples and templates. Template values were used for constituents below the detection limits for sample data or constituents not measured from the sampling event. Templates are based on sampling data from tanks that contain the same waste type, supplemented with Revision 5 of the HDW model data (RPP-19822). Table 5 provides the best basis inventory concentration estimate for Np^{237}, Pu^{238}, Pu^{239}, Pu^{240}, and Am^{241} in the sludges stored in these tanks. These five radionuclides comprise the majority of the transuranic elements with half-lives greater than 20-years present in these wastes. In general, the concentrations of Np^{237}, Pu^{238}, Pu^{239} and Pu^{240}, in the sludges stored in tanks 241-B-203, 241-B-204 and 241-T-201 through 241-T-204 are calculated from the analyzed total alpha concentrations for these wastes using an alpha isotope distribution template.

Core samples of the wastes in tanks 241-B-201 and 241-B-202 were obtained in 1991. Core samples of the wastes in tanks 241-B-203 and 241-B-204 were obtained in 1995. Core samples of the wastes in tanks 241-T-201 through 241-T-204 were obtained in 1997. The sludges collected in these core samples were analyzed to determine the composition of these wastes as well as the concentration of alpha emitting radionuclides (gross alpha analysis).

The analyzed, mean gross alpha analyses, 95% lower confidence limit and 95% upper confidence limit for the wastes in tanks 241-B-201 through 241-B-204 and tanks 241-T-201 through 241-T-204 are provided in Table 6 (Wilmarth 2002). Uranium-238, which is not a transuranic element, would be included in the gross alpha analysis. The gross alpha analysis would tend to over estimate the sum of the concentrations of alpha-emitting transuranic isotopes with half-life greater than 20 years. This is substantiated by the analyses of the core samples from tank 241-B-201 and 241-B-202, which were also analyzed to determine the concentrations of Pu^{238}, Pu^{239}, Pu^{240}, and Am^{241} in these sludges (see Table 5). The sums of the concentrations of alpha-emitting transuranic isotopes with half-life greater than 20 years are ~829ηCi/g and ~218ηCi/g, respectively for tanks 241-B-201 and 241-B-202.

The analyzed gross alpha analyses for the waste in each tank are in excess of 100ηCi/g. The sum of the concentrations of Np^{237}, Pu^{238}, Pu^{239}, Pu^{240}, and Am^{241} for the sludges in tanks 241-B-201 and 241-B-202 also indicate that the concentration of transuranic elements with half-life greater than 20-years is also in excess of 100ηCi/g.

Table 5. Transuranic Element Analytical Results for Sludges

Tank Name	Analyte	Basis	Concentration, µCi/g
241-B-201	237Np	TE	1.45E-07
241-B-201	238Pu	S	6.29E-03
241-B-201	239Pu	S	7.51E-01
241-B-201	240Pu	S	4.32E-02
241-B-201	241Am	S	2.84E-02
241-B-202	237Np	TE	2.26E-07
241-B-202	238Pu	S	2.03E-03
241-B-202	239Pu	S	1.28E-01
241-B-202	240Pu	S	2.17E-02
241-B-202	241Am	S	6.67E-02
241-B-203	237Np	S	8.17E-07
241-B-203	238Pu	C	2.02E-03
241-B-203	239Pu	C	2.34E-01
241-B-203	240Pu	C	2.94E-02
241-B-203	241Am	S	3.46E-02
241-B-204	237Np	TE	2.08E-07
241-B-204	238Pu	C	1.72E-03
241-B-204	239Pu	C	1.99E-01
241-B-204	240Pu	C	2.51E-02
241-B-204	241Am	C	3.84E-02
241-T-201	237Np	TE	1.37E-07
241-T-201	238Pu	C	2.26E-03
241-T-201	239Pu	C	6.69E-01
241-T-201	240Pu	C	4.51E-02
241-T-201	241Am	C	3.96E-02
241-T-202	237Np	TE	2.20E-07
241-T-202	238Pu	C	1.45E-03
241-T-202	239Pu	C	1.67E-01
241-T-202	240Pu	C	2.11E-02
241-T-202	241Am	C	3.23E-02
241-T-203	237Np	S	6.20E-07
241-T-203	238Pu	C	1.73E-03
241-T-203	239Pu	C	2.00E-01
241-T-203	240Pu	C	2.53E-02
241-T-203	241Am	S	3.36E-02
241-T-204	237Np	S	5.35E-07
241-T-204	238Pu	C	1.46E-03
241-T-204	239Pu	C	1.68E-01
241-T-204	240Pu	C	2.12E-02
241-T-204	241Am	S	2.01E-02

Notes: Radionuclides are decay corrected to January 1, 2004
S – Sample based
C – Calculated
TE – Based on a Hanford Defined Waste model or engineering based waste template

Table 6. Gross Alpha Analyses for Sludges.				
Tank	Mean (ηCi/g)	Relative Standard Deviation	95% Low Confidence Limit	95% Upper Confidence Limit
B-201	1,310	13%	1,030	1,590
B-202	398	9%	338	457
B-203	215	9%	184	245
B-204	265	9%	226	303
T-201	757	21%	490	1,024
T-202	223	12%	180	265
T-203	196	12%	157	234
T-204	144	10%	120	169

5.0 SUMMARY

Tanks 241-B-201 through 241-B-204 received waste from the plutonium concentration activities conducted from October 1946 through June 1952 in the 224-B Concentration building. Following cessation of spent nuclear fuel processing activities in the 221-B Bismuth Phosphate Plant and cleanout of the plant inventory, tanks 241-B-201 through 241-B-204 received flush solutions from equipment cleanout in the 221-B and 224-B buildings and metal waste transfer lines.

Tanks 241-T-201 received waste from plutonium concentration activities conducted from November 4, 1946 through May 24, 1949 in the 224-T Concentration building. Tanks 241-T-202 through 241-T-204 received wastes from plutonium concentration activities conducted from May 24, 1949 through May 29, 1952 in the 224-T Concentration building.

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APPENDIX A

WASTE STATUS SUMMARY REPORTS

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[redacted] Security Information

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WASTE STATUS SUMMARY

Period: April, May and June, 1952

Classification Cancelled and Changed To
Declassified
By Authority of DOC 1973
By A.E. Barker
Verified By J.P. Deraint
Bill Hanna
APPROVED FOR 3/15/96
PUBLIC RELEASE

SPECIAL RE-REVIEW
FINAL DETERMINATION
DECLASSIFICATION CONFIRMED
BY Bill Hanna DATE Aug 16, 1995
BY J.D. Brown DATE 8/17/95
Pat Duncan 3-15-96
J.K. Rice 3-22-96

A.C. Morganthaler
Planning & Scheduling Group
Waste Control
Manufacturing Department

INSPECTED BY
R.M. Allen 7-29-02
NAME DATE
PNNL NSAT

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WASTE STATUS SUMMARY
SEPARATIONS-OPERATIONS

0 02

Month of April, 1952

	B Plant	T Plant
Runs of Metal Waste to waste farm during the month	7 7	59
Gals. increase in Metal Waste volume	22,688	165,000
Gals./Run average Metal Waste volume	2,[illegible]	2,797
Reserve Gals. available for Metal Waste on 4/30/52	1,693	1,906
Runs of First Cycle Waste sent to farm during the month	7	66
Gals. increase in First Cycle Waste volume	22,000	160,000
Gals./Run average First Cycle Waste volume	3,112	2,424
Reserve Gals. available for First Cycle Waste on 4/30/52	3,131	2,444

Remarks: * Average computed after correcting line 02 for 2 [illegible] runs and 1 Bicarbonate flush (1244 & 957)

	S Plant
Batches equivalent to 6e3 MS product content contributing to make-up of waste	115.96
Gallons increase in fission product waste storage*	324,500
Tons of Uranium (as-charged) processed to contribute to make-up of waste	83.14
Gallons/batch equivalent to 6e3 MS	2,315
Gallons per ton of Uranium processed	3,900
Reserve Gallons available for Redox waste on 4/30/52	8,272

Remarks: * During the month of February a 21% discrepancy between amount of waste sent versus amount of waste received in storage existed. The variation this month between these two figures was less than 1%.

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0 03

WASTE STORAGE STATUS

As of 2/30/72

METAL WASTE

FARM	Gallons x 10³ In Storage Begg Mos	Metal Waste Received	In Storage End Mos	Reserve Capacity In Batches
EAST AREA				
B	1579		1579	
C	3374		3374	
BX	3117		3117	23
BY	4389	23	4412	604
TOTAL	12459	23	12482	627
WEST AREA				
T	1579		1579	
U	4737	(1579)	3158	
TX	5489	165	5654	132
TY				558**
TOTAL	11805	165	10391	790

*REMARKS: *Reserve capacity at 2700* Gallons per batche

** The present plan is to allocate 101-102TY to Redox reserves and give equivalent space in U farm to T plant for metal waste storage. This would eliminate the necessity of completing the removal facilities in TY farm. (Removal facilities are not available in U farm). This plan requires that a cascade become available in U farm before 106TX, the present T plant metal waste tank, is filled.

( ) Metal waste in 101-102-103 U slurrying for TXP feed.

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WASTE STORAGE STATUS

[illegible] 1/26/[illegible]

URANIUM CONTENT
METAL WASTE

Tank	TONS			
	In Storage Beg. Mos	Received	To TBP	In Storage End Mos
EAST AREA				
B	[illegible]			[illegible]
C	[illegible]			[illegible]
IX	[illegible]			[illegible]
IT	[illegible]			[illegible]
TOTAL	[illegible]			[illegible]
WEST AREA				
T	375			375
U	1,227			[illegible]
TX	1,260	[illegible]		772
TY				[illegible]
TOTAL	[illegible]	[illegible]		[illegible]

* Tons of uranium transferred as metal waste to Sludge Handling Group for slurrying.

REMARKS: April [illegible] based on [illegible] is based on 600 [illegible]

H-[illegible] 4-52

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WASTE STORAGE STATUS

0 05

As of 4/30/52

FIRST CYCLE WASTE

FARM	Gallons x 10³				
	In Storage Beg. Mo.	1st Cycle Waste Recd	Feed To Evaporator	In Storage End Mo.	* Reserve Cap. In Batches
EAST AREA					
B	3,060			3,060	
Evaporator Feed Tank - 106 B	233		54	187	123
C	3,039		496	2,543	221
BX	3,175			3,175	
BY	2,088	22		2,108	58
TOTAL	9,593	22	642	9,073	405
WEST AREA					
T	3,910			3,910	
U	3,585		687	898	
TX	3,176	160		3,336	374
Evaporator Feed Tank - 118 TX	69		280	49	116
TY					
TOTAL	4,740	160	203	4,493	1,020

* Reserve Capacity at 2,800 gallons per batch.
) Not included in totals.

REMARKS:

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0 06

WASTE STORAGE STATUS

As of 6/30/52

SECOND CYCLE WASTES

	Cascade In Use	Gallons x 10³ Cribbed	Gallons x 10³ Sludge	Estimated Reserve Capacity In Batches
EAST AREA	110-111-112-[illegible]	100	461 **	3,065
WEST AREA	110-111-112-[illegible]	24	666	3,140
TOTAL		124	1,067	6,205

REMARKS: * Inventory calculated on basis of [illegible] sludge/run combined 5-6 to 2nd cycle only.

** Intbated sludge on 5-1-52.

224 BUILDING WASTES

	Cascade In Use	Gallons x 10³ Cribbed	Gallons x 10³ Sludge	Estimated Reserve Capacity In Batches
EAST AREA	204-203-[illegible]	35	166	512 **
WEST AREA	203-202-[illegible]	273	181	255 runs*
TOTAL		201	348	767

REMARKS: * Reserve capacity = 0.4 sludge/run = based on sludge reading 5/4/52. 145 Reserve runs in 200,203,2[illegible] [illegible] 111 reserve runs in 201-Tg

** Reserve capacity = [illegible] gal/run = based on 1/2/52 readings = 232 runs in tanks 204, 201, 202 and 220 [illegible]

*** Basis = 190 gal/inch of sludge

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WASTE STORAGE STATUS

0 07

As of 4/30/52

TBP WASTE

FACP	TBP Gallons x 10³ In Storage Beg. Mo.	Total Waste Received	In Storage End Mo.	Reserve Capacity
EAST AREA				
B				
C				
IX				
TP				
TOTAL				
WEST AREA				
T				9,263
U				216
TX				
TP				3,110
TOTAL				4,589

a Supernate tanks 209 BY and 115 TX now included in the above.

Gallons x 10³ of condensate to grib _____

Ratios * Feed to Waste

During Month _____

Cumulative to date _____

* Feed adjusted to metal waste equivalent

REMARKS: 216,000 gal. in 202-U series allocated to TBP wastes

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[redacted] STE STORAGE STATUS

DECLASSIFIED As of 4/30/52

WASTE EVAPORATOR

FARM	Gallons x 10^3 Feed	Bottoms	Condensate To Cribs	Reserve Bottoms Space
EAST AREA				
B		122		654
C	442			
BX				
BY				
TOTAL	442	122	320	654
WEST AREA				
T				
U	407			
TX		112		466
TY				
TOTAL	407	112	295	466

REMARKS:

REDOX WASTE

FARM	Gallons x 10^3 In Storage Beg. Mo.	Waste Received	In Storage End Mo.	** Reserve Capacity In Batches
S Farm	356	325	681	2,970
*U Farm				
TOTAL	356	325	681	2,970

* Gallons x 10^3 of condensate to crib 320 ?

* Will be considered as Redox reserve capacity when empty.

** Reserve Capacity at 2,815 gallons per 6-3 equivalent batch based on current month's performance.

REMARKS:

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STATUS OF WASTE FARMS

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	* Type of Waste	Gallons x 10^3 Capacity	Gallons x 10^3 Stored	Reserve Cap. Gallons x 10^3	Reserve Cap. In Batches	Remarks
B Farm						
101	M	530	530			
102	M	530	530			
103	M	530	529			
104	1C	530	530			
105	1C	530	530			
106	1C	530	187	343	123	1st Cycle Evaporator & Feed Tank
107	2C	530	220	310		
108	2C	530	500	29		
109	2C	530	10	515		Butt-a tank & not filling
110	2C	530	530			
111	2C	530	530			Active Cascade B Farm
112	2C	542	542			
201	224	54.5	54.5			Cascading to crib
202	224	54.5	54.5			
203	224	54.5	54.5			Active Cascade 224-B
204	224	54.5	54.5			Cascading 204, 203, 2C to crib
						Idle
C Farm						
101	M	530	530			
102	M	530	530			
103	M	530	519			
104	M	530	530			
105	M	530	530			
106	M	530	519			
107	1C	530	399	131	47	
108	1C	530	34	496	177	
109	1C	530	525			Finished pumping 4-15-52
110	1C	530	530			
111	1C	530	530			
112	1C	530	525			
201	M	54.5	54.5			
202	M	54.5	54.5			
203	M	54.5	54.5			
204	M	54.5	52.5			

* M - Metal Waste
1C - First Cycle Waste
2C - Second Cycle Waste
R - Redox Waste

224 = 224 Bldg. Waste
TBP = TBP Waste
EB = Evaporator Bottoms

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STATUS OF WASTE FARMS

[redacted] 16

	Type of Waste	Gallons x 10^3 Capacity	Gallons x 10^3 Stored	Reserve Capacity Gallons x 10^3	Reserve Capacity In Batches	Remarks
BX Farm						
101	[illegible]	530	530			
102	[illegible]	530	467	63	23	
103	[illegible]	530	530			
104	[illegible]	530	530			
105	[illegible]	530	530			
106	[illegible]	530	530			
107	1C	530	530			
108	1C	530	530			
109	[illegible]	530	530			
110	[illegible]	530	530			
111	1C	530	[illegible]			
112	1C	530	530			
BY Farm						
101	[illegible]	758	758			
102	[illegible]	758	758			
103	[illegible]	758	661	83	31	
104	[illegible]	758	758			
105	[illegible]	758	491	267	99	
106	[illegible]	758	[illegible]	347	272	
107	2C	758	759			
108	2C	758	758			
109	TBP	758				Supernate tank
110	1C	758	597	161	[illegible]	Sole tank now filling @ H Plant
111	[illegible]	758	758			
112	[illegible]	758	225	531	197	W1 tank now filling @ S Plant
[illegible] Farm						
101	[illegible]	530	530			
102	[illegible]	530	530			
103	[illegible]	530	518			
104	1C	530	530			
105	1C	530	530			
106	2C	530	526			
107	1C	530	245	285		
108	1C	530	73	457		Reserved for TBP
109	1C	530	[illegible]	523		

* MW - Metal Waste
1C - First Cycle Waste
2C - Second Cycle Waste
S - Sweet Waste
[illegible]

224 = 224 Bldg. Waste
TBP = TBP Waste
EB = Evaporator Bottoms

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STATUS OF WASTE FARMS

0 10

	* Type of Waste	Gallons x 10³ Capacity	Gallons x 10³ Stored	Reserve Capacity Gallons x 10³	Reserve Capacity In Batches	Remarks
T Farm						
110	2C	530	530			
111	2C	530	530			Active cascade @ T Plant
112	2C	569	569			Cascading to crib
201	224	54.5	54.5			
202	224	54.5	54.5			
203	224	54.5	54.5			
204	224	54.5	54.5			Cascading to crib Idle
U Farm						
101	MW	530	530			
102	MW	530	530			
103	MW	530	519			Starting for feed to TSF
104	MW	530	530			
105	MW	530	530			
106	MW	530	519			
107	MW	530	530			
108	MW	530	530			
109	MW	530	519			
110	1C	530	136	194		
111	1C	530	530			Down to sludge 4/28/52
112	1C	530	32	498		Pump now installed here Stopped pumping 4/24/52
201	TBP	54.5		54.5		
202	TBP	54.5		54.5		Plan over-ground filling with TBP waste
203	TBP	54.5		54.5		
204	TBP	54.5		54.5		
X Farm						
101	MW	758	758			
102	MW	758	758			
103	MW	758	758			
104	MW	758	750			
105	MW	758	758			
106	MW	758	758			
107	MW	758	758			
108	MW	758	356	391	137	MW tank now filling - T Plant
109	1C	758	574	184	66	
110	1C	758	9	749	268	1-C tank now filling - T Plant
111	1C	758		758	272	
112	1C	758		758	272	

* MW - Metal Waste
* 1C - First Cycle Waste
2C - Second Cycle
R - Redox Waste
T - [illegible]

224 - 224 Bldg. Waste
TBP - TBP Waste
EB - Evaporator Bottoms

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STATUS OF WASTE TANKS

0 12

	Type of Waste	Gallons x 10^3 Capacity	Gallons x 10^3 Stored	Reserve Capacity Gallons x 10^3	Reserve Capacity In Batches	Remarks
TX Farm (contd)						
113	[illegible]	758	[illegible]	166		
114	[illegible]	758	[illegible]			
115	[illegible]	[illegible]				
116		758	[illegible]			Active - TF supernate
117	[illegible]	758	[illegible]			
118	[illegible]	758	[illegible]			
TY Farm						
101		758				
102		[illegible]		[illegible]	[illegible]	
103		758				
104		758				
105		758				
106	[illegible]	758				
107						
S Farm						
101	[illegible]	758			#	
102	[illegible]	758		[illegible]	[illegible]	
103		758		[illegible]	[illegible]	
104		758		[illegible]	[illegible]	
105	[illegible]	758		[illegible]	[illegible]	
106	[illegible]	758		[illegible]	[illegible]	
107		758		[illegible]	[illegible]	
108	[illegible]	758		[illegible]	[illegible]	
109	[illegible]	758		[illegible]	[illegible]	
110		758		[illegible]	[illegible]	
111	[illegible]	758		[illegible]	[illegible]	
112	[illegible]	758		[illegible]	[illegible]	

MW = Metal Waste
1C = First Cycle Waste
2C = Second Cycle Waste
R = Redox Waste

224 = 224 Bismuth Waste
TBP = TBP Waste
SD = Supernatant Decants

[illegible] Reserve capacity based on current month's [illegible] average

[illegible]

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WASTE STATUS SUMMARY

Month of May, 1952

	B Plant	T Plant
Runs of Metal Waste to waste farm during the month	6	21
Gals. increase in Metal Waste volume	13,063	55,000
Gals./Run average Metal Waste volume	2,172	2,619
Reserve Gals. available for Metal Waste on 5-30-52	3,669,000	3,842,000
Runs of First Cycle Waste sent to farm during the month	6	18
Gals. average per First Cycle Waste volume	22,000	43,250
Gals./Run average First Cycle Waste volume	3,663	2,403
Reserve Gals. available for First Cycle Waste on 5-30-52	2,487,000	2,625,000

Remarks: * After correction for 2 BX runs sent to storage during this time, the average volume per run is 2,221 gallons.

	U Plant
Bismuth Phosphate Batch Equivalent	2[illegible],56
Gallons increase in fission product waste storage	342,530
Tons of Uranium, (as charged), processed, to contribute to make-up of waste	28.73
Gallons/batch equivalent to 6-3 MS	2,742
Gallons per ton of Uranium processed	4,351
Reserve Gallons available for Redox waste on 6-1-52	8,045,000

Remarks: * Last month this figure was reported at 3,999 gal/ton, on 5-19-52 an error was found in the calculated amount of uranium charged to extraction, which gave a corrected figure of 4,687 gal/ton. This month's figure of 4,351 gal/ton is based on preliminary uranium inventory figures.

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WASTE STORAGE STATUS

As of 5-31-52

METAL WASTE

TANK	Gallons x 10³ In Storage Beg Mo	Metal Waste Received	In Storage End Mo	Reserve Capacity In Batches*
EAST AREA				
B	1,579		1,579	
C	3,374		3,374	
UR	3,337		3,337	23
TW	4,492	33	4,525	595
TOTAL	12,782	33	12,815	618
WEST AREA				
T	1,579		1,579	
U	4,737	(1,579)	4,737	
TX	5,654	55	5,709	120
TY	MT		MT	558
TOTAL	11,970	55	12,025	682

REMARKS: *Reserve capacity at 2,700 Gallons per batch

( ) Metal waste in 101-102-103-U slurrying for TBP feed

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WASTE STORAGE STATUS

As of 5-31-52

URANIUM CONTENT
METAL WASTE

FARM	TONS PURE URANIUM In Storage Beg. Mo.	Received	* To TBP	In Storage End Mo.
EAST AREA				
B	385			385
C	843			843
BX	762			762
BY	1,059	5		1,064
TOTAL	3,847	5		3,852
WEST AREA				
T	375			375
U	772			772
TX	1,297	11		1,308
TY				
TOTAL	2,444	11		2,455

* Tons of uranium transferred as metal waste to the Waste Removal Group for slurrying.

REMARKS: ** Metal in tank 101-102-103 U equal to 405 tons has been credited to TBP. This does not show as uranium present in totals.

May increase based on 0.622 tons of uranium per run. This is based on 600 MWD/ton average.

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WASTE [illegible] STATUS

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As of 5-21-52

FIRST CYCLE WASTE

TANK	Gallons x 10³ In Storage Beta Flow	Gallons x 10³ Last Cycle Waste Recd	Gallons x 10³ Feed To Evaporator	Gallons x 10³ In Storage Alpha Flow	Percent Reserve Capacity
EAST AREA					
B **	2,060			1,060	
Evaporator Feed Tank - 106 B	187		38	225	320
C	2,543		425	2,118	376
UX	3,175			3,175	
UY	2,108	22		2,130	88
TOTAL	9,073	22	388	8,707	584
WEST AREA					
T	1,910			1,910	
U	898		516	382	
TX **	1,336	41		1,377	859
Evaporator Feed Tank - 118 TX	349		187	536	79
TY					
TOTAL	4,493	41	329	4,205	938

* Reserve Capacity at 2,800 gallons per batch
( ) Not included in totals

REMARKS: ** Excluding 106-B and 118-TX evaporator feed tanks.

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WASTE STORAGE STATUS ~~0-35~~ 17

As of 5-31-52

SECOND CYCLE WASTES

	Cascade In Use	Gallons x 10^3		Estimated Reserve Capacity In Batches
		Cribbed	Sludge	
EAST AREA	110-111-112 S	21	103	3,059
WEST AREA	110-111-112 T	36	672	2,454***
TOTAL		57	1,071	

REMARKS: ** Reserve calculations based on 225 gal/run combined 5-6 and 2nd cycle only, or 275 gal/run and 75 gallons on cleaning per run from 224 Bldg.

*** Assuming 224 wastes with 2nd cycle and 5-6 waste to give a total sludge volume of 350 gal/run.

224 BUILDING WASTES

	Cascade In Use	Gallons x 10^3		Estimated Reserve Capacity In Batches
		Cribbed	Sludge	
EAST AREA	204-203-202 S	26	166	506
WEST AREA 5-29	203-202 T 110-111-112 T	215	187	Combined with 2nd cycle above
TOTAL		241	353	506

REMARKS: On 5-29-52, 224-T wastes were tied into 5-6 stream feeding to 110-111-112 T at diversion box 241-152-T. Thus in 200 West Area 5-6, 2nd cycle, and 224 wastes are all combined in these tanks.

H-2902-05-X 4-52

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[redacted] STORAGE STATUS

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As of 5-31-52

TBP WASTE

FARM	Gallons x 10^3			
	In Storage Beg. Mo.	TBP Received	In Storage End Mo.	Reserve Capacity
EAST AREA				
B				
C				
BX				
BY				
TOTAL				
WEST AREA				
T				
U		8		1,263
TX				216
TY				
TOTAL				3,110
				8,589

Supernate tanks, 109 BY and 115 TX not included in the above.

Gallons x 10^3 of condensate to crib _____

Ratio: * Feed to Waste.

During Month _____

Cumulative to date _____

* Feed adjusted to metal waste equivalent.

REMARKS: Present plan is to store cold-start up wastes of 10 g/liter and under in 200-U tanks; if over 10 g/liter to send it to 115-TX supernatant tank for reworking.

For hot TBP wastes: if 10 g/l or less to discard to current TBP waste storage tanks; if between 10-32 g/l (up to 25% of metal content in feed) to be discarded into T Plant current NI cascade.

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As of 5-31-52

WASTE EVAPORATOR

FARM	Gallons x 10^3 Feed	Bottoms	Condensate To Crib	Reserve Bottoms Space
EAST AREA				
B		93		761
C	388			
BX				
BY				
TOTAL	388	93	295	761
108-B filled with bottoms 5-14-52. 109-B into service 5-15-52.				
WEST AREA				
T				
U	329			
TX		96		370
TY				
TOTAL	329	96	233	370

REMARKS: 110,111,112-U tanks are not emptied to sludge; tanks remaining to evaporate in 200 West area are 114-TX and cascade 104,105,106-T which filled 3-31-52.

REDOX WASTE

FARM	Gallons x 10^3 In Storage Beg. Mo.	Waste Received	In Storage End Mo.	** Reserve Capacity In Batches
S Farm	681	343	1,024	2,905
SY Farm				
TOTAL	681	343	1,024	2,905

Gallons x 10^3 of condensate to crib 387

* 4119 to be considered as Redox reserve capacity when empty.

** Reserve Capacity at 2,744 gallons per 6-3 equivalent batch

REMARKS: 110-3 tank cascaded to 111-S on 5-9-52.

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STATUS OF WASTE TANKS

	Type of Waste	Gallons x 10^3 Capacity	Gallons x 10^3 Stored	Reserve Cap. Gallons x 10^3	Reserve Cap. In Batches	Remarks
B Farm						
101	MW	530	530			
102	MW	530	530			
103	MW	530	519			
104	1C	530	530			
105	1C	530	530			
106	1C	530	225	300	120	1st Cycle Evaporator - Feed Tank
107	EB	530	220	310		
108	EB	530	530			
109	EB	530	74	451		Bottoms tank - now filling
110	2C	530	530			Active Cascade - B Plant
111	2C	530	530			
112	2C	542	542			Cascading to crib
201	224	54.5	54.5			Active Cascade - 224-B
202	224	54.5	54.5			
203	224	54.5	54.5			Cascading 204,201,202 to crib
204	224	54.5	54.5			Idle
C Farm						
101	MW	530	530			
102	MW	530	530			
103	MW	530	519			
104	MW	530	530			
105	MW	530	530			
106	MW	530	519			
107	1C	530	399	131	47	
108	1C	530	34	496	177	
109	1C	530	525			
110	1C	530	530			
111	1C	530	530			
112	1C	530	99	426	152	Tank - last current
201	MW	54.5	54.5			
202	MW	54.5	54.5			
203	MW	54.5	54.5			
204	MW	54.5	54.5			

MW - Metal Waste
1C - First Cycle Waste
2C - Second Cycle Waste
R - Redox Waste
224 - 224 Bldg. Waste
TBP - TBP Waste
EB - Evaporator Bottoms

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STATUS OF WASTE TANKS

[redacted]

	* Type of Waste	Gallons x 10^3		Reserve Capacity		Remarks
		Capacity	Stored	Gallons x 10^3	In Batches	
BX Farm						
101	MW	530	530			
102	MW	530	467	63	23	
103	MW	530	530			
104	MW	530	530			
105	MW	530	530			
106	MW	530	530			
107	1C	530	530			
108	1C	530	530			
109	1C	530	530			
110	1C	530	530			
111	1C	534	525			
112	1C	530	530			
BY Farm						
101	MW	758	758			
102	MW	758	758			
103	MW	758	664	83	31	
104	MW	758	758			
105	MW	758	491	267	99	
106	MW	758		747	277	
107	1C	758	758			
108	1C	758	753			
109	TBP	758				Supernate tank
110	1C	758	619	224	10	[illegible] tank now filling at B Plant
111	MW	758	758			
112	MW	758	238	509	188	MW tank now filling at B Plant.
T Farm						
101	MW	530	530			
102	MW	530	530			
103	MW	530	519			
104	1C	530	530			
105	1C	530	530			
106	1C	530	528			
107	1C	530	245	285		Reserved for TBP
108	1C	530	73	457		
109	1C	530	4	521		

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* MW - Metal Waste
1C - First Cycle Waste
2C - Second Cycle Waste
R - Redox Waste
224 - 224 Bldg. Waste
TBP - TBP Waste
ES - Evaporator Bottoms

[redacted]

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[redacted]

	* Type of Waste	Gallons x 10^3 Capacity	Gallons x 10^3 Stored	Reserve Capacity Gallons x 10^3	Reserve Capacity In Batches	Remarks
Farm						
110	2C	530	530			Active cascade - T Plant
111	2C	530	530			
112	2C	560	560			Cascading to tank
201	22L	54.5	54.5			These 4 tanks out of service 5-29-[illegible]
202	22L	54.5	54.5			
203	22L	54.5	54.5			
204	22L	54.5	54.5			
Farm						
101	MM	530	530			
102	MM	[illegible]	530			Sitting for feed to TBP
103	MM	[illegible]	510			
104	MM	530	530			
105	MM	530	530			
106	MM	530	510			
107	MM	530	530			
108	MM	530	530			
109	MM	530	510			
110	1C	530	336	194		
111	1C	530	14	516		
112	1C	530	32	498		
201	TBP	54.5		32.5		Plan eventual filling with TBP waste
202	TBP	54.5		54.5		" " "
203	TBP	54.5		54.5		" " "
204	TBP	54.5		54.5		" " "
X Farm						
101	MM	758	758			
102	MM	758	758			
103	MM	758	758			
104	MM	758	758			
105	MM	758	758			
106	MM	758	758			
107	MM	758	758			
108	MM	758	413	[illegible]	[illegible]	[illegible] - T Plant
109	2C	758	625	[illegible]	[illegible]	[illegible] - T Plant
110	2C	758	[illegible]	[illegible]	[illegible]	
111	2C	758	[illegible]	[illegible]	[illegible]	
112	2C	758	[illegible]	[illegible]	[illegible]	

* MM - Metal waste
1C - First Cycle Waste
2C - Second Cycle
R - Redox waste
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[illegible] - [illegible]
[illegible] - [illegible]
[illegible] - [illegible]

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STATUS OF WASTE TANKS

	Type of Waste	Gallons x 10^3 Capacity	Gallons x 10^3 Staged	Reserve Capacity Gallons x 10^3	Reserve Capacity In Batches	Remarks
TX Farm (cont'd)						
113	EB	758	388	370	-	
114	IC	758	752			
115	TBP					Active - TBP supernate
116	EB	758	758			
117	EB	758	756			
118	IC	758	536	222	79	Evaporator feed tank
TY Farm						
101	MW	758		758	281	
102	MW	758		747	272	
103	TBP	758		758		
104	TBP	758		747		
105	TBP	758		758		
106	TBP	758		747		
S Farm						
101	R	758		758	** 277	
102	R	758		758	277	
103	R	758		747	273	
104	R	758		758	277	
105	R	758		758	277	
106	R	758		747	273	
107	R	758		758	277	
108	R	758		758	277	
109	R	758		747	273	
110	R	758	774		-	
111	R	758	249	509	186	Active tank
112	R	758		747	273	

* MW = Metal Waste
IC = First Cycle Waste
SC = Second Cycle Waste
R = Redox Waste
224 = 224 Sludge Waste
TBP = TBP Waste
EB = Evaporator Bottoms

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Month of June, 1952

	B Plant	T Plant
Runs of Metal Waste to waste farm during the month	5	16
Gals. increase in Metal Waste volume	13,750	49,500
Gals./Run average Metal Waste volume	2,292	3,094
Reserve Gals. available for Metal Waste on 7-1-52	1,655,000	1,791,000
Runs of First Cycle Waste sent to farm during the month	7	17
Gals. increase in First Cycle Waste volume	19,250	57,750
Gals./Run average First Cycle Waste volume	2,750	3,357
Reserve Gals. available for First Cycle Waste on 7-1-52	1,719,000	2,889,000

Remarks: Higher average volume per run on 1st cycle and MW for T Plant, in part due to increased volumes for processing I.P. runs.

	S Plant
Bismuth Phosphate Batch Equivalent	90.51
Gallons increase in fission product waste storage	264,000
Tons of Uranium, (as charged), processed, to contribute to [illegible] of waste	54.84
Gallons/batch equivalent to 6-3 MS	* 2,917
Gallons per ton of Uranium processed	* 4,814
Reserve Gallons available for Redox waste on 7-1-52	7,781,000

Remarks: * The increase in waste volumes for the month of June was principally the result of abnormal uranium rework.

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H-2902.01-X 3-52

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HW-27[illegible]

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WASTE STORAGE STATUS

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[redacted]

METAL WASTE

FARM	Gallons x 10^3 In Storage Beg. Mo.	Metal Waste Received	In Storage End Mo.	Reserve Capacity In Batches
EAST AREA				
B	1,579		1,579	
C	3,374		3,374	
BX	3,117		3,117	23
BY	4,425	14	4,439	590
TOTAL	12,495	14	12,509	613
WEST AREA				
T	1,579		1,579	
U	4,737	(2,579)	4,737	
TX	5,709	50	5,759	106
TY	MT	-	MT	558
TOTAL	12,025	50	12,075	664

REMARKS: *Reserve capacity at 2,700 Gallons per batch.

Metal waste in 101,102,103-U slurrying for TBP feed.

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WASTE STORAGE STATUS

[redacted] as of 6-30-52

URANIUM CONTENT
METAL WASTE

FARM	TONS Uranium In Storage Beg. Mo.	Received	* To TBP	In Storage End Mo.
EAST AREA				
B	385			385
C	841			841
BX	762			762
BY	1,064	6		1,070
TOTAL	3,052	6		3,058
WEST AREA				
T	375			375
U	772		**	772
TX	1,308	13 & 5***		1,326
TY	-		-	-
TOTAL	2,455	13		2,473

* Tons of uranium transferred as metal waste to the Waste Removal Group for slurrying.

REMARKS: ** Metal in tanks 101-102-103 U equal to 405 tons of uranium has been processed to TBP. This does not show as uranium present in totals.

June increase is for all runs having June serial numbers not necessarily in storage at month end.

*** Cold uranium sent from TBP Building to 115 TX (see TB2 waste).

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[redacted]ASIN STORAGE STATUS

[redacted] as of 6-30-52

[redacted]

FACILITY	Gallons x 10^3				*
	In Storage Beg. Mo.	1st Cycle Waste Rec'd	Feed To Evaporator	In Storage End Mo.	Reserve Capacity In Batches
EAST AREA					
B **	1,060	-	-	1,060	-
Evaporator Feed Tank - 106 B	225	-	32	193	120
[illegible]	2,117	-	214	1,903	452
TX	3,175	-	-	3,175	-
DT	2,130	19	-	2,149	41
TOTAL	8,707	19	246	8,480	613
WEST AREA					
T	1,910	-	-	1,910	-
U	382	-	-	382	-
TX **	1,377	58	195	1,240	838
Evaporator Feed Tank - 118 TX	536	-	331	205	198
TX	-	-	-	-	-
TOTAL	4,205	58	526	3,737	1,036

* Reserve Capacity at 2,800 gallons per batch

( ) Not included in totals.

Remarks: ** Excluding 106-B and 118-TX evaporator feed tanks. Tank 113-TX containing 753,000 gal. 1st cycle waste will be designated as Evaporator Bottoms Storage Tank when emptied.

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WASTE STORAGE STATUS

[redacted] of 6-30-52

[redacted]

	Cascade In Use	Gallons x 10^3		Estimated Reserve Capacity In Batches
		Cribbed	Sludge	
EAST AREA	110-111-112-B	15	407	3,044 **
WEST AREA	110-111-112-T	39	685	2,415 ***
TOTAL		54		

REMARKS: ** Reserve calculations based on storage of combined 5-6 and 2nd cycle only at 275 gallons per run.

*** Reserve calculation based on storage of combined 5-6, 2nd cycle, and 224 wastes at 350 gallons per run.

224 BUILDING WASTES

	Cascade In Use	Gallons x 10^3		Estimated Reserve Capacity In Batches
		Cribbed	Sludge	
EAST AREA	204-203-202-B	40	167	500
WEST AREA	204-201-T 110-111-112-T	Inactive Cascade 86	187 + Sludge for 110-111-T shown above.	
TOTAL		126	354	500

REMARKS:

[redacted]

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WASTE STORAGE STATUS

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[redacted] As of 6-30-52

[redacted]

FARM	Gallons x 10^3 In Storage Beg. Mo.	TBP Received	In Storage End Mo.	Reserve Capacity
EAST AREA				
B				
C				
BY				
BY				
TOTAL				
WEST AREA				
T				1,263
U				216
TX				
TX				3,110
TOTAL				4,589

Supernate tanks, 109 BY and 115 TX not included in the above.

Gallons x 10^3 of condensate to crib (216-W) - This calculation is not complete - will be available for next month's report.

Feed to Plants

During Month --

Cumulative to date --

* Feed adjusted to metal waste equivalent

REMARKS: 10,991# cold uranium sent to 115-TX tank equal to 193,992 gallons received in 115-TX.

28,746 gallons of waste to ditch equivalent to 149.8 # cold uranium.

[redacted]

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[redacted] 6-30-52

WASTE EVAPORATOR

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FARM	Gallons x 10^3 Feed	Bottoms	Condensate To Cribs	Reserve Bottoms Space
EAST AREA				
B		98	148	663
C	246			
BX				
BY				
TOTAL	246	98	148	663
WEST AREA				
T				
U				
TX	526	144	382	421
TY				
TOTAL	526	144	382	421*

* 114-TX now pumping; reclaimed space designated as Evap. bottoms reserve space.

REMARKS: 109-C now pumping in East Area.

REDOX WASTE

FARM	Gallons x 10^3 In Storage Beg. Mo.	Waste Received	In Storage End Mo.	Reserve Capacity In Batches
S Farm	1,024	264	1,288	2,668
W Farm	-	-	-	
TOTAL	1,024	264	1,288	2,668

Gallons x 10^3 of condensate to crib 292.

* Will be considered as Redox reserve capacity when empty.

** Reserve Capacity at 2,917 gallons per 6-3 equivalent batch

REMARKS: [redacted]

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[redacted]

R-2962.07-X 4-52

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[redacted]

	Type of Waste	Gallons x 10^3 Capacity	Gallons x 10^3 Stored	Reserve Cap. Gallons x 10^3	Reserve Cap. In Batches	Remarks
B Farm						
101	MW	530	530			
102	MW	530	530			
103	MW	530	519			
104	1 C	530	530			
105	1 C	530	530			
106	1 C	530	193	337	120	1st Cycle evaporator - Feed tank.
107	EB	530	220	310	-	
108	EB	530	530	-	-	
109	EB	530	172	353	-	Bottom tank - now filling.
110	2 C	530	530			Active cascade - B Plant.
111	2 C	530	530			
112	2 C	542	542			Cascading to crib.
201	224	54.5	54.5			Active cascade - 224-B.
202	224	54.5	54.5			
203	224	54.5	54.5			Cascading 204, 203, 202 to crib.
204	224	54.5	54.5			Idle
C Farm						
101	MW	530	530			
102	MW	530	530			
103	MW	530	519			
104	MW	530	530			
105	MW	530	530			
106	MW	530	519			
107	-	530	399	131	47	
108	1 C	530	34	495	377	
109	1 C	530	311	214	76	Tank - now pumping to 106-B.
110	1 C	530	530			To be reserved for TBP supernate.
111	1 C	530	530			
112	1 C	530	99	425	152	Partially pumped to be finished at a later date.
[illegible]	MW	54.5	54.5			
[illegible]	MW	54.5	54.5			
[illegible]	MW	54.5	54.5			
[illegible]	MW	54.5	54.5			

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MW - Metal Waste
1 C - First Cycle Waste
2 C - Second Cycle Waste
EB - Redox Waste

224 - 224 Bldg. Waste

[redacted]

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[redacted]

HW-27838

STATUS OF WASTE FARMS

[redacted] 0-32

	Type of Waste	Gallons x 10³ Capacity	Gallons x 10³ Stored	Reserve Capacity Gallons x 10³	Reserve Capacity In Cascade	Remarks
HX Farm						
101	MW	530	530			
102	MW	530	457	63	23	
103	MW	530	530			
104	MW	530	530			
105	MW	530	530			
106	MW	530	530			
107	1 C	530	530			
108	1 C	530	530			
109	1 C	530	530			
110	1 C	530	530			
111	1 C	530	525			
112	1 C	530	530			
BY Farm						
101	MW	758	758			
102	MW	758	758			
103	MW	758	664	83	31	
104	MW	758	758			
105	MW	758	491	267	99	
106	MW	758	-	747	277	
107	1 C	758	758			
108	1 C	758	753			
109	TBP	758				Supernate tank.
110	1 C	758	638	115	41	1-C tank now filling - B Plant.
111	MW	758	758			
112	MW	758	252	495	183	MW tank now filling - B Plant.
T Farm						
101	MW	530	530			
102	MW	530	530			
103	MW	530	519			
104	1 C	530	530			Cascade filled 3-31-52. Next and last cascade of 1st cycle waste to evaporate in West area. To take place after sufficient allowance for aging.
105	1 C	530	530			
106	1 C	530	528			
107	1 C	530	245			TBP - waste - active cascade.
108	1 C	530	73			
109	1 C	530	4			

MW - Metal Waste
1C - First Cycle Waste
2C - Second Cycle Waste
R - Redox Waste

224 - 224 Bldg. Waste

[redacted] Evaporator Bottoms

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STATUS OF WASTE FARMS

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	Type of Waste	Gallons x 10^3 Capacity	Gallons x 10^3 Stored	Reserve Capacity Gallons x 10^3	Reserve Capacity In Batches	Remarks
T Farm						
110	224 and 2 C	530	530			Active cascade - T Plant.
111	224 and 2 C	530	530			
112	224 and 2 C	569	569			Cascading to crib.
201	224	54.5	54.5			
202	224	54.5	54.5			
203	224	54.5	54.5			
204	224	54.5	54.5			
U Farm						
101	MW	530	530			Cascading - now slurrying for feed to TBP plant.
102	MW	530	530			
103	MW	530	519			
104	MW	530	530			
105	MW	530	530			
106	MW	530	519			
107	MW	530	530			
108	MW	530	530			
[illegible]	MW	530	519			
110	1 C	530	336	494		Cascade being held as Redox reserve space.
111	1 C	530	14	516		
112	1 C	530	32	498		
201	TBP	54.5		52.5		Plan overhead filling with TBP waste.
202	TBP	54.5		54.5		
203	TBP	54.5		54.5		
204	TBP	54.5		54.5		
TX Farm						
101	MW	[illegible]	758			
102	MW	758	758			
103	MW	758	758			
104	MW	758	750			
105	MW	758	758			Cascade set up to receive high TBP wastes in addition to MW from T plant.
106	MW	758	758			
107	MW	758	758			
108	MW	758	461	206	105	MW tank now filling via 107 TX - T plant.
109	1 C	758	673	85	30	1-C tank now filling T plant.
110	1 C	758	9	749	268	
[illegible]	1 C	758	[illegible]	758	271	
[illegible]	1 C	758	[illegible]	[illegible]	[illegible]	

* MW - Metal Waste.
* 1C - First Cycle Waste
* 2C - Second Cycle
* R - Redox Waste
* TBP - TBP Waste
* EB - Evaporator Bottoms

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[redacted] 34

[redacted] STATUS OF WASTE TANKS

[redacted]

	Type of Waste	Gallons x 10^3 Capacity	Gallons x 10^3 Stored	Reserve Capacity Gallons x 10^3	Reserve Capacity In Batches	Remarks
TX Farm (cont'd)						
113	EB	758	532	226	-	Active bottoms tank - West area.
114	1 C	758	558	195	-	Transfer pumping - reclaimed space to
115	TBP					active TBP supernate; to be designated
116	EB	758	758	-		as EB.
117	EB	758	756	-		
118	1 C	758	205	553	198	Evaporator feed tank.
TY Farm						
101	MW	758		750	201	
102	MW	758		747	277	
103	TBP	758		758	-	
104	TBP	758		747	-	
105	TBP	758		758	-	
106	TBP	758		747	-	
S Farm						
101	R	758	-	758	260	
102	R	758	-	758	260	
103	R	758	-	747	256	
104	R	758	-	758	260	
105	R	758	-	758	260	
106	R	758	-	747	256	
107		758	-	758	260	
108	R	758	-	758	260	
109	R	758	-	747	256	
110	R	758	774	-		
111	R	758	513	245	84	Active tank.
112	R	758	-	747	256	

* MW - Metal Waste
1C - First Cycle Waste
2C - Second Cycle Waste
R - Redox Waste
224 - 224 Bldg. Waste
TBP - TBP Waste
EB - Evaporator Bottoms

** Reserve figures based on current months average number of gallons per batch equivalent 2,917 gal/batch.

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WASTE STATUS SUMMARY

Period: JULY, AUGUST AND SEPTEMBER 1952

Classification Cancelled and Changed To
DECLASSIFIED
By Authority of DOE 1973
By A.E. Bollen 5-10-82
Verified By W.P. DeFoan

A.C. Morganthaler
Planning & Scheduling Group
Waste Control
Manufacturing Department
SEPARATIONS SECTION

SPECIAL RE-REVIEW
FINAL DETERMINATION
DECLASSIFICATION CONFIRMED
BY Bill Hanson DATE Aug 19, 1995
BY J.A. Beggs DATE 8/17/95
Pat Linen 3-15-96
LK Rice 3-15-96

3-15-96
Bill Hanson
APPROVED FOR
PUBLIC RELEASE

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INSPECTED BY
Ron Iten 7-29-02
NAME DATE
PNNL NSAT

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Month of July, 1952

	B Plant	T Plant
Runs of Metal Waste to waste farm during the month	8	38
Calls. Increase in Metal Waste volume	12,250	(1) 101,500
Gals./Run average Metal Waste volume	2,406	(2) 2,705
Reserve Gals. available for Metal Waste on 8-1-52	1,636,000	1,687,000
Runs of First Cycle Waste sent to farm during the month	8	36
Gals. increase in First Cycle Waste volume	15,125	122,000
Gals./Run average First Cycle Waste volume	1,891	(3) 3,388
Reserve Gals. available for First Cycle Waste on 8-1-52	(4) 243,000	3,487,000

Remarks: (1) Includes 1720 gallons of uranium waste from Bldg. 321 - 300 Area. See (uranium waste).
(2) From T Plant process only.
(3) High average volume per run due in part to increased volumes resulting from processing L.P. runs.
(4) Drop in 1st cycle reserve capacity for 200-E Area due to reallocation of space in tank farm 241-C to TBP waste reserve system.

	S Plant
Bismuth Phosphate Batch Equivalent	128.42
Gallons increase in fission product waste storage	345,280
Tons of Uranium (as charged), processed, to contribute to make-up of waste	77.2
Gallons/batch equivalent to 6-3 MS	8,688
Gallons per ton of Uranium processed	4,473
Reserve Gallons available for Redox waste on 8-1-52	3,422,000

Remarks: Includes only space available in tank farm 241-C.

[redacted]

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[redacted] WASTE STORAGE STATUS

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As of 7-31-52

METAL WASTE

FORM	Gallons x 10^3 In Storage Beg. Mo.	Metal Waste Received	In Storage End Mo.	Reserve Capacity In Batches
EAST AREA				
B	1,579		1,579	
C	2,374		2,374	
BX	3,117		3,117	23
BY	4,439	19	4,453	[illegible]
TOTAL	12,509	19	12,521	[illegible]
WEST AREA				
T	1,579		1,579	
U	4,737	(1,579)	4,737	
TX	5,759	105	5,864	67
TY	[illegible]		[illegible]	[illegible]
TOTAL	12,075	105	12,180	625

REMARKS: *Reserve capacity at 2,700 Gallons per batch.

( ) [illegible] in cascade 2-1-4-10-11-12-13 [illegible] c & x [illegible] pool.

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WASTE STORAGE STATUS

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As of 7-31-52

URANIUM CONTENT
METAL WASTE

FARM	TONS DEPLETED URANIUM			
	In Storage Beg. Mo.	Received	* To TBP	In Storage End Mo.
EAST AREA				
B	383			383
C	837			837
HX	758			758
BY	1,064.93	3.50	(1) -10.67	1,057.76
TOTAL (5)	3,042.93	3.50	-10.67	3,035.76
WEST AREA				
T	374			374
U	770			770
TX	1,322.08	24.97 (2)(3)		1,347.05
TY				
TOTAL	2,466.08	24.97		2,491.05 (4)

* Tons of uranium transferred as metal waste to the Waste Removal Group for slurrying.

REMARKS: (1) Adjustment for loss of 10.67 tons uranium via overflow of 102 BY in Feb., 1951.
(2) Includes 0.41 tons of unreacted metal received on 7-2-52 into 108-TX tank from Bldg. 321 - 300 Area.
(3) Includes 1.77 tons of depleted and decontaminated metal into 115-TX tank from TBP.
(4) Total does not include metal in tank 101-102-103-U now sluicing for feed to TBP Plant.
(5) All uranium storage figures have been adjusted to equal those kept by S.F. Accountability group.

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WASTE STORAGE STATUS

As of 7-31-52

FIRST CYCLE WASTE
AND CONTROL WASTE
TANK FARMS

FARM	Gallons x 10^3				
	In Storage Beg Mos	1st Cycle Waste Rec'd	Feed To Evaporator	In Storage End Mos	* Reserve Cap In Batches
EAST AREA					
B **	1,060			1,060	-
Evaporator Feed Tank - 101 B	193		(-) 199	392	49
C	1,903		(-) 845	1,058	To TBP reserve
BX	3,175			3,175	
BY	2,140	15		2,161	38
TOTAL	8,450	25	646	7,846	87
WEST AREA					
T	1,710			1,710	
U	382			382	
TX **	1,240	122	555	807	1,063
Evaporator Feed Tank - 101 TX	205		(-) 33	244	184
TY					
TOTAL	3,737	122	516	3,343	1,247

* Reserve Capacity at 1,000 gallons per batch
( ) Not included in total

REMARKS: ** Including evaporator feed tanks.

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WASTE STORAGE STATUS

[redacted] 7-31-52

SECOND CYCLE WASTES

	Cascade In Use	Gallons x 10^3		Estimated Reserve Capacity In Batches
		Cribbed	Sludge	
EAST AREA	110-111-112-E	21	409	3,036 **
WEST AREA	110-111-112-C	93	698	2,377 ***
TOTAL		114	1,107	5,413

REMARKS: ** Reserve calculations based on storage of combined 5-6 and 2nd cycle only at 275 gallons/row.

*** Reserve calculations based on storage of combined 5-6, 2nd cycle, and 224 wastes at 350 gallons/row.

224 BUILDING WASTES

	Cascade In Use	Gallons x 10^3		Estimated Reserve Capacity In Batches
		Cribbed	Sludge	
EAST AREA	201-203-202-E	24	168	402
WEST AREA	201-201-T 110-111-112-T	In active cascade 116	187* Sludge & waste from with 2nd cycle.	
TOTAL		140	355	402

REMARKS:

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WASTE STORAGE STATUS

As of 7-31-52

[redacted]

TBP WASTE

FARM	Gallons x 10^3 In Storage Beg. Mo.	TBP Waste Received	In Storage End Mo.	Reserve Capacity
EAST AREA				
B				
C				(3) 2,100
BX				
BY				
TOTAL				2,100
WEST AREA				
T				1,263
U				216
TX		(1) 177		
TY				3,110
TOTAL				4,589

Separate tanks, 109 BY and 115 TX not included in the above.

Gallons x 10^3 of condensate to crib (216-W) 605

Ratio: * Feed to Waste

During Month _____

Cumulative to date _____

* Feed adjusted to metal waste equivalent.

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REMARKS: (1) Waste to 115-TX = 177 x 10^3 gallons equivalent to 1.77 tons of decontaminated and depleted uranium.

(2) Waste to ditch 52 x 10^3 gallons = 1.06 tons of decontaminated and depleted uranium.

(3) Space released by East Area evaporator Concentrate Bldg. C 107,8,9, and 113,11,12 [redacted]

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[redacted]

[redacted]

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As of 7-31-[illegible]

[redacted]

FARM	Gallons x 10^3 Feed	Bottoms	Condensate To Cribs	Reserve Bottoms Space
EAST AREA				
B		223	425	442
C	616			
BY				
SY				
TOTAL	616	223	425	442
WEST AREA				
T				
U				
TX	516	143	373	268
TY				
TOTAL	516	143	373	268

[redacted]

REDOX WASTE

FARM	Gallons x 10^3 In Storage Beg. Mo.	Waste Received	In Storage End Mo.	Reserve Capacity In Batches
S Farm	1,287	345	1,632	2,766
W Farm				
TOTAL	1,287	345	1,632	2,766

Gallons x 10^3 of condensate to crib 360

Will be considered as Redox reserve capacity when empty.

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Reserve Capacity at 2,688 gallons [redacted]

REMARKS: Additional storage space equal to 1,200,000 gallons is available for Redox waste in tanks 110-111 [redacted]

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[redacted]

	530		
	530		
	519		
	530		
	530		
	572	138	(after dumping batch E-1 8-1-52) Evap. feed tank.
	220	310	
	530		
	573	137	Active bottom tank - Feed from evap.
	530		Active cascade - B. Flank
	530		
	502		Tank 112-B cascade to crib.
	54.5		Active cascade - 216-B
	54.5		
	54.5		Tank 202-B flow to crib.
	54.5		Tank 216-B to [illegible]
	530		
	530		
	519		
	530		
	530		
	519		
	399	131	Space allocated to TBP waste reserve
	34	166	"
	17	502	" finished pumping 7-25-52.
			finished pump
	231	299	Space allocated to TBP, inc 7-22-52.
	237	293	Tank pumping at north end; to be held
	140	319	Space allocated to TBP. for TBP.
	54.5		
	54.5		
	54.5		
	[redacted]		

[illegible] Waste
[illegible] Waste
[illegible] Waste
[illegible] Waste

224 - [illegible] Bldg. Waste

[redacted]

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STATUS

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	Type of Waste	Gallons Capacity	Gallons Stored	Capacity Gallons x 10^3	Capacity In Batches	Remarks
B Farm						
101	1W	530	530			
102	1W	530	467	63	23	
103	1W	530	530			
104	1W	530	530			
105	1W	530	530			
106	1W	530	530			
107	1C	530	530			
108	1C	530	530			
109	1C	530	530			
110	1C	530	530			
111	1C	530	525			
112	1C	530	530			
B Farm						
101	1W	758	758			
102	1W	758	758			
103	1W	758	664	63	31	
104	1W	758	758			
105	1W	758	491	267	99	
106	1W	758	-	747	277	
107	1C	758	758			
108	1C	758	753			
109	TBP	758				TBP supernate tank - 200 East Area
110	2C	758	653	105	38	1-C tank now filling - B Plant
111	1W	758	758			
112	1W	758	271	476	176	1W tank now filling - B Plant
I Farm						
101	1W	530	530			
102	1W	530	530			
103	1W	530	519			
104	1C	530	530			Content of cascade will be aged 1
105	1C	530	530			years 3-31-53
106	1C	530	528			
107	1C	530	245	235	-	Space allocated to TBP waste
[illegible]	1C	530	73	457	-	
[illegible]	1C	530	4	[illegible]		

[redacted]

1W - Waste
1C - First Cycle Waste
2C - Second Cycle Waste
[illegible] - Waste

224 - 224 Bldg. Waste
TBP - TBP Waste

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[redacted]

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[redacted]

STATUS OF WASTE FARMS

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	Type of Waste	Gallons x 10³ Capacity	Stored	Gallons x 10³	In Patches	Remarks
T Farm						
109	224-2C	530	530			Active cascade - T Plant
111	224-2C	530	530			
113	224-2C	569	519			113-C cascade to grib
201	224	54.5	54.5			
202	224	54.5	54.5			
203	224	54.5	54.5			
204	224	54.5	54.5			
U Farm						
101	[illegible]	530	530			Cascade now slurrying for food to TBP Plant.
102	[illegible]	530	530			
103	[illegible]	[illegible]	519			
104	[illegible]	530	530			
105	[illegible]	530	530			
106	[illegible]	530	519			
107	[illegible]	530	530			
[illegible]	[illegible]	530	530			
109	[illegible]	530	519			
110	1C	530	336	194		Cascade being held at Tolex waste, pending.
111	1C	530	14	516		
112	1C	530	32	498		
201	TBP	54.5	-	52.5		Plan overland filling with TBP waste.
202	TBP	54.5	-	54.5		
203	TBP	54.5	-	54.5		
204	TBP	54.5	-	54.5		
TX Farm						
101	1C	758	758			
102	1C	758	758			
103	1C	758	758			
104	1C	758	750			
105	1C	758	758			
106	1C	758	758			Cascade now receiving high TBP wastes in addition to T Plant metal waste.
107	1C	758	758			
108	1C	758	556	101	67	Active M1 tank - T Plant
109	1C	758	758			Tank filled on Run 7-12-67-Refill 7-25-5
[illegible]	1C	758	45	713	255	
110	1C	758	[redacted]			
112	1C	758				

* MW - Metal Waste
* 1C - First Cycle Waste
2C - Second Cycle Waste

224 - 224 Bldg. Waste

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[redacted]

[redacted]

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STATUS OF WASTE FARMS

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	* Type of Waste	Gallons x 10^3 Capacity	Gallons x 10^3 Stored	Reserve Capacity Gallons x 10^3	Reserve Capacity In Batches	Remarks
TX Farm (cont'd)						
113	EB	758	675	83		Active bottoms tank - east Area.
114	1C	758	4	749	268	pumping stopped 7-11-52
115	TBP					active TBP supernate
116	RR	758	758			
117	RR	758	756			
118	1C	758	244	514	184	Evaporator feed tank
TY Farm						
101	[illegible]	758		758	261	
102	[illegible]	758		747	277	
103	TBP	758		758		
104	TBP	758		747		
105	TBP	758		758		
106	TBP	758		747		
S Farm						
101	R	758		758	282 **	
102	R	758		758	282	
103	R	758		747	278	
104	R	758		758	282	
105	R	758		758	282	
106	R	758		747	278	
107	R	758		758	282	
108	R	758		758	282	
109	R	758		747	278	
110	R	758	754			
111	R	758	755			Tank capacity to 112-S 7-25-52
112	R	758	103	644	240	

* MW = Metal Waste
1C = First Cycle Waste
2C = Second Cycle Waste
R = Redox Waste

224 = 224 Bldg. Waste
TBP = TBP Waste
EB = Evaporator Bottoms

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** Reserve figures for storage in tank [redacted] on Redox current [redacted] 2,688 gallons/batch.

N-2902.11-X 4-52

[redacted]

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[redacted]

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WASTE STATUS SUMMARY

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[redacted]

Month of August 1952

	B Plant	T Plant
Runs of Metal Waste to waste farm during the month	1 run - 4 AW	46
Gals. increase in Metal Waste volume	20,625	132,000
Gals./Run average Metal Waste volume	--	2,870
Reserve Gals. available for Metal Waste on 9-1-52	1,615,000	784,000
Runs of First Cycle Waste sent to farm during the month	1 run - 5½ AW	49
Gals. increase in First Cycle Waste volume	26,125	110,000
Gals./Run average First Cycle Waste volume	--	2,245
Reserve Gals. available for First Cycle Waste on 9-1-52	202,000	2,791,000

Remarks: (1) Tank 241-TX-114 added to T plant MW reserve.

(2) Averages not calculated for B plant since they no longer have production significance.

	S Plant
Bismuth Phosphate Batch Equivalent	166.76
Gallons increase in fission product waste storage	405,780
Tons of Uranium, (as charged), processed, to contribute to make-up of waste	109.4
Gallons/batch equivalent to 6-3 MS	2433 (1)
Gallons per ton of Uranium processed	3709
Reserve Gallons available for Redox waste on 9-1-52	(2) 7,027,000

Remarks: (1) This figure represents a 12% variation from average plant performance which cannot be entirely accounted for by August process changes.

(2) Includes reserve in 5 Farnsworth-Hochman lines are available at present [redacted] allocated reserves in U and TX farms.

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[redacted] DECLASSIFIED 27839 p.14.

[redacted]

[redacted] As of 8-31-52

METAL WASTE

FARM	Gallons x 10^3 In Storage Beg. Mo.	Metal Waste Received	In Storage End Mo.	Reserve Capacity In Batches
EAST AREA				
B	1572		1572	
C	3374		3374	
[illegible]	3117		3117	23
BY	4458	21	4479	575
TOTAL	12520		12542	598
WEST AREA				
T	1579		1579	
U	4737 (a)		4737	
TX	5868	741 (2)	6605 (1)	200 (3)
TY	Mt		Mt	None (4)
TOTAL	12180		12921	200

REMARKS: *Reserve capacity at 2700 gallons per batch.

(1) MW in cascade 101-102-103-U slurrying for TBP feed.

(2) Includes 7000 gallons received from TBP plant.

(3) 241-TX-106 tank pumping overground at month end to tank 111TX. Metal waste reserve increase due to addition of 111TX to metal waste account.

(4) Reserve space in tanks 241-TY-101 and 2 removed from metal waste account and transferred to Redox reserves because of incomplete [illegible] facilities in these tanks.

[redacted]

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[redacted]

WASTE STORAGE STATUS

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As of 8-31-52

[redacted]

URANIUM CONTENT
METAL WASTE

FARM	TONS In Storage Beg. Mo.	Received	* To TBP	In Storage End. Mo.
EAST AREA				
B	383			383
C	837			837
BX	758			758
BY	1057.76	--	--	1057.76
TOTAL	3035.76			3035.76
WEST AREA				
T	374			374
U	770			770
TX	1347.05	29.66 (1)		1376.71
TY				
TOTAL	2491.05	29.66		2520.71 (2)

* Tons of uranium transferred as metal waste to the Waste Removal Group for slurrying.

REMARKS: (1) Includes 1.51 tons from TBP (decontaminated and depleted).

(2) Total does not include metal in tanks 241-U-101-102-103 now sluicing for feed to TBP plant (405 tons)

[redacted]

[redacted]

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WASTE STORAGE STATUS

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As of 8-31-52

[redacted]

URANIUM CONTENT
METAL WASTE

FARM	TONS			
	In Storage Beg. Mo.	Received	* To TBP	In Storage End Mo.
EAST AREA				
B	383			383
C	837			837
BX	758			758
BY	1057.76	--	--	1057.76
TOTAL	3035.76			3035.76
WEST AREA				
T	374			374
U	770			770
TX	1347.05	29.66 (1)		1376.71
TY	--			
TOTAL	2491.05	29.66		2520.71 (2)

* Tons of uranium transferred as metal waste to the Waste Removal Group for slurrying.

REMARKS: (1) Includes 1.51 tons from TBP (decontaminated and depleted)

(2) Total does not include metal in tanks 241-U-101-102-103 now sluicing for feed to TBP plant (405 tons)

[redacted]

[redacted]

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[redacted] STATUS

[redacted] 8-31-52

[redacted]

FIRST CYCLE WASTE
AND COATING WASTE
T AND B PLANTS

FARM	Gallons x 10^3				*
	In Storage Beg. Mo.	1st Cycle Waste Rec'd	Feed To Evaporator	In Storage End Mo.	Reserve Cap. In Batches
EAST AREA					
B **	1060		-309	751	--
Evaporator Feed Tank - 106 B	392		10	402	46
C	1058		-31	727	--
TX	3175			[illegible]	--
BY	2164	26		2190	26
TOTAL	7849	26	-630	7245	72
WEST AREA					
T	1910			1910	
U	382			382	
TX **	807	110		917	755 (1)
Evaporator Feed Tank - 118 TX	244		-154	90	242
TY	--			--	
TOTAL	3343	110	-154	3299	997

* Reserve Capacity at 2800 gallons per batch
( ) Not included in totals.

REMARKS: ** Excluding evaporator feed tanks.

(1) Tank 241-TX-114 has been converted to metal waste storage account causing a loss of 268 batches to 1st cycle waste reserve (see Metal waste storage).

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[redacted]

WASTE STORAGE STATUS

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8-21-57

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SECOND CYCLE WASTES

	Cascade In Use	Gallons x 10^3		Estimated Reserve Capacity In Batches
		Cribbed	Sludge	
EAST AREA	110-111-112-B	21	411	3030 **
WEST AREA	110-111-112-T	99	715	2328 ***
TOTAL		120	1126	5358

REMARKS: ** Reserve calculations based on storage of combined 5-6 and 2nd cycle only at 275 gallons of sludge per run.

*** Reserve calculation based on storage of combined 5-6, 2nd cycle, and 224 wastes at 350 gallons of sludge per run.

224 BUILDING WASTES

	Cascade In Use	Gallons x 10^3		Estimated Reserve Capacity In Batches
		Cribbed	Sludge	
EAST AREA	204-203-202-B	29	169	486
WEST AREA	204-201-T 110-111-112-T	Inactive cascade 155	187 sludge and reserve	-- shown with 2nd cycle
TOTAL		184	356	486

REMARKS:

[redacted]

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WASTE STORAGE STATUS [redacted]

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As of 8-31-52

[redacted]

TBP WASTE

FARM	Gallons x10³ In Storage Beg. Mo.	TBP Waste Received	In Storage End Mo.	Reserve Capacity
EAST AREA				
B				
C				2443 (4)
BX				
BY				
TOTAL				2443
WEST AREA				
T				1263
U				216
TX		29 (3)		
TY				3110
TOTAL				4589

Supernate tanks, 109 BY and 115 TX not included in the above.

Gallons x 10³ of condensate to crib 565 (2168 W) *(1)*(2)

Ratio: * Feed to Waste

During Month _____

Cumulative to date _____

* Feed adjusted to metal waste equivalent

REMARKS: (1) Includes 0.79 tons of depleted and decontaminated uranium.

(2) In addition to the amount cribbed, 242,000 gallons containing 0.79 tons of decontaminated and depleted uranium were ditched.

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(3) 9000 gallons of waste were sent [redacted] active metal waste cascade. This waste contained 1.51 tons of decontaminated and depleted Uranium.

(4) Reserve space has increased during m[illegible] 1st cycle evaporation.

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WASTE STORAGE STATUS

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As of 6-31-52

W[redacted]

[redacted]

FARM	Gallons x 10^3 Feed	Bottoms	Condensate To Crubs	Reserve Bottoms Space
EAST AREA				
B	299	203	427	548 (1)
C	331			
BX				
BY				
TOTAL	640	203	427	
WEST AREA				
T				
U				
TX	154	45	102	38 (2)
TY				
TOTAL	154	45	102	

REMARKS: (1) Bottoms reserve has increased due to pumping of 105-B, a future bottoms tank.
(2) Re-evaporation of bottoms tanks 116-117 and 113 TX to begin in near future to obtain additional bottoms reserve space.

REDOX WASTE

FARM	Gallons x 10^3 In Storage Beg. Mo.	Waste Received	In Storage End Mo.	Reserve Capacity In Batches
S Farm	1632	406	2038	2558 (1)
*U Farm				
TOTAL	1632			2558

Gallons x 10^3 of condensate to trib 381

* Will be considered as Redox reserve capacity when [redacted]

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** Reserve Capacity at 2750 (2) gallons per 6-3 equivalent batch

REMARKS: (1) Additional Redox [redacted] in tanks 241-110, 111, 112 and 241-TY-101-102 [redacted] 208,000 and 1,505,000 gallons respectively.
(2) [redacted] average of [redacted]

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[redacted]

STATUS OF WASTE TANKS

[redacted]

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	** Type of Waste	Gallons x 10^3 Capacity	Gallons x 10^3 Stored	[redacted] x 10^3	[redacted] Batches	Remarks
B Farm						
101	MW	530	530			
102	MW	530	530			
103	MW	530	519			
104	1C	530	530			
105	1C	530	221	309	E.B.	Tank now being pumped to 106-B
106	1C	530	402	128	46	Evap. feed tank
107	EB	530	298	232		Active bottoms tank-East Area Evap.
108	EB	530	530			
109	EB	530	518	7	[illegible]	Aug. 21, 1952 - switched to 107-B
110	2C	530	530			
111	2C	530	530			Active cascade - B plant
112	2C	542	542			
201	224	54.5	54.5			Tank 102-B cascades to crib
202	224	54.5	54.5			Active cascade - 224-B
203	224	54.5	54.5			
204	224	54.5	54.5			202-B flows to crib
						204-B is idle
C Farm						
101	MW	530	530			
102	MW	530	530			
103	MW	530	519			
104	MW	530	530			
105	MW	530	530			
106	MW	530	519			
107	1C	530	399	131		All space in tanks 107-112C
108	1C	530	34	496		allocated to TBP waste reserve
109	1C	530	10	515		
110	1C	530	231	299		
111	1C	530	36	494		Finished pumping to 106-B 8-2-52
112	1C	530	17	518		Finished pumping to 106-B 8-15-52
201	MW	54.5	54.5			
202	MW	54.5	54.5			
203	MW	54.5	54.5			
204	MW	54.5	54.5			

[redacted]

MW - Metal Waste
[illegible] - First Cycle Waste
[illegible] - Second Cycle Waste
[illegible] - Redox Waste

224 - 224 Bldg. Waste
TBP - TBP Waste
EB - Evaporator [redacted]

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H-2902.08-A 4-52

[redacted]

[redacted]

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	Type of Waste*	Gallons x 10^3 Capacity	Gallons x 10^3 Stored	Res[redacted]apacity Gallons x 10^3	Res[redacted]apacity In	Remarks
BX Farm						
101	MW	530	530			
102	MW	530	467	63	23	
103	MW	530	530			
104	MW	530	530			
105	MW	530	530			
106	MW	530	530			
107	1C	530	530			
108	1C	530	530			
109	1C	530	530			
110	1C	530	530			
111	1C	530	525			
112	1C	530	530			
BY Farm						
101	MW	758	758			
102	MW	758	758			
103	MW	758	664	83	31	
104	MW	758	758			
105	MW	758	491	267	99	
106	MW	758	--	747	277	
107	1C	758	758			
108	1C	758	753			
109	TBP	758		758		TBP supernate tank 200 East Area
110	1C	758	679	76	26	
111	MW	758	758			
112	MW	758	292	455	168	
T Farm						
101	MW	530	530			
102	MW	530	530			
103	MW	530	519			
104	1C	530	530			
105	1C	530	530			contents of cascade will be aged [redacted] 3-31-53, after which time [redacted]oration will be started
106	1C	530	528			
107	1C	530	245	285		space allocated to TBP waste
108	1C	530	73	457		
109	1C	530	4	515	[redacted]	

* MW - Metal Waste
1C - First Cycle Waste
2C - Second Cycle Waste
R - Redox Waste
224 - 224 Bldg. Waste
TBP - TBP Waste
EB - Evaporator Bottoms

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STATUS OF WASTE TANKS [redacted]

	Type of Waste*	Gallons x 10^3 Capacity	Gallons x 10^3 Stored	Reserve Capacity Gallons x 10^3	Reserve Capacity In Batches	Remarks
T Farm						
110	224 & 20	530	530			Active cascade at T Plant
111	224 & 20	530	530			
112	224 & 20	569	569			112-T cascades to crib
201	224	54.5	54.5			
202	224	54.5	54.5			
203	224	54.5	54.5			
204	224	54.5	54.5			
U Farm						
101	MW	530	530			Cascade now slurrying for feed to TBP Plant
102	MW	530	530			
103	MW	530	519			
104	MW	530	530			
105	MW	530	530			
106	MW	530	519			
107	MW	530	530			
108	MW	530	530			
109	MW	530	519			
110	1C	530	376	194	71**	Cascade being held as Redox waste reserve
111	1C	530	14	516	187	
112	1C	530	32	498	181	
201	TBP	54.5	--	54.5		Plan overhead filling with TBP waste
202	TBP	54.5	--	54.5		
203	TBP	54.5	--	54.5		
204	TBP	54.5	--	54.5		
TX Farm						
101	MW	758	758			
102	MW	758	758			
103	MW	758	758			
104	MW	758	758			
105	MW	758	758			
106	MW	758	758			
107	MW	758	758			
108	MW	755	697	60	18	[redacted] 114 TK on 5-29-52
109	1C	758	758	--	--	
110	1C	758	1[illegible]	[illegible]	215	Tank [illegible] filling T plant 1st cycle
111	1C	758	--	758	[illegible]	
112	1C	758	--	[illegible]	[illegible]	

* MW - Metal Waste
* 1C - First Cycle Waste
20 - Second Cycle
R - Redox Waste
224 - 224 Indg. Waste
TBP - TBP Waste
ER - [redacted]

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[redacted]

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STATUS OF WASTE FARMS

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[redacted]

	Type of Waste*	Gallons x 10^3 Capacity	Gallons x 10^3 Stored	Reserve Capacity Gallons x 10^3	Reserve Capacity In Batches	Remarks
TX Farm (cont'd)						
119	MB	758	720	38	--	
114	1C & MW	758	13	734	272	started filling 8-29-52 from 106 TX [illegible]
115	TBP					
116	EB	758	758			
117	EB	758	756			
118	1C	758	90	578	242	Evap.
SX Farm						
201	R	758		758	**	Space reserve transferred to Redox account - Tank 101 and 102
202	R	758		747	272	
203	TBP	758		758		
204	TBP	758		747		
205	TBP	758		758		
[illegible]	TBP	758		747		
S Farm						
101	R	758		758	**	
102	R	758		758	276	
103	R	758		747	272	
104	R	758		758	276	
105	R	758		758	276	
106	R	758		747	272	
107	R	758	67	621	351	tank started filling 8-25-52
108	R	758		758	276	
109	R	758		747	272	
110	R	758	774	--	--	
111	R	758	755	--	--	
112	R	758	442	305	111	cascade abandoned on 8-25-52 because of boiling in tank 110-S

* MW = Metal Waste
1C = First Cycle Waste
2C = Second Cycle Waste
R = Redox Waste

224 [redacted]
TBP - TBP Waste
EB - Evaporator Bottoms

** Reserve fi[redacted]e in tank farm 241-S and tanks 101[redacted] on average plant performance of 2750 gallons per batch.

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WASTE STATUS SUMMARY

[redacted]

[redacted] of September 1952

	B Plant	T Plant
Runs of Metal Waste to waste farm during the month	1 water flush	8
Gals. increase in Metal Waste volume	2060	20600
Gals./Run average Metal Waste volume	- (1)	2575
Reserve Gals. available for Metal Waste on 10/1/52	1,613,000	773,000
Runs of First Cycle Waste sent to farm during the month	10-1/2 AW	5 & 1 AW
Gals. increase in First Cycle Waste volume	43,300	22,000
Gals./Run average First Cycle Waste volume	- (1)	3550 (2)
Reserve Gals. available for First Cycle Waste on 10/1/52	113,000	2,114,000

Remarks: (1) Averages not calculated for B Plant since runs are not of normal size.

(2) T Plant first cycle waste average calculated after correcting for volume of acid wash.

	S Plant
Bismuth Phosphate Batch Equivalent	156.41
Gallons increase in fission product waste storage	390,500
Tons of Uranium, (as charged), processed to contribute to make-up of waste	96.72
Gallons/batch equivalent to 5-3 [illegible]	2497
Gallons per ton of Uranium processed	4037
Reserve Gallons available for Redox waste on 10/1/52 (3)	6,636,000

Remarks: (3) [illegible] S Farm only: 2,713,000 gallons of reserve space are immediately [illegible] if a tie line is made between Redox system and remainder of farm system.

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[redacted]

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[redacted]

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WASTE STORAGE STATUS

As of 9/30/52

METAL WASTE

FARM	Gallons x 10³ In Storage Beg. Mo.	Metal Waste Received	In Storage End Mo.	Reserve Capacity In Batches
EAST AREA				
B	1579		1579	
C	3374		3374	
BX	3117		3117	23
BY	4479	2	4481	574
TOTAL	12549	2	12551	597
WEST AREA				
T	1579		1579	
U	4737 (1)		4737	
TX	6005	21 (-7) (2)	6019	287
TY	MT		MT	None
TOTAL	12321	14	12335	287

REMARKS: *Reserve capacity at 2700 Gallons per batch.

(1) Metal waste in cascade 241-U-101-102-103 slurrying for TBP feed.

(2) By transferring metal waste from 241-TX-106 to 241-TX-114 an apparent volume loss of 7,000 gallons occurred. This can be accounted for by the increase of density resulting from cooling liquid in the 106-TX tank prior to pumping to the 114-TX tank.

[redacted]

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[redacted]

[redacted]

[redacted]

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WASTE STORAGE STATUS

As of 9/30/52

URANIUM CONTENT
METAL WASTE

FARM	Tons In Storage Begg No.	Received	* To TBP	In Storage End No.
EAST AREA				
B	383			383
C	837			837
[redacted]	758			758
[redacted]	1057.76			1057.76
TOTAL	3035.76			3035.76
WEST AREA				
T	374			374
U	770			770
[redacted]	1376.71	4.92		1381.63
[redacted]				
TOTAL	2520.71	4.92		2525.63(1)

* Tons of uranium transferred as metal waste to the Waste Removal Group for slurrying.

REMARKS: (1) 405 tons of Uranium in cascade 241-U-101-102-103 now slurrying for feed to TBP Plant not included in this total.

[redacted]

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WASTE STORAGE STATUS

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AS OF 9/30/52

FIRST CYCLE WASTE AND COATING WASTE
T AND B PLANTS

FARM	Gallons x 10^3				
	In Storage Beg. Mo.	1st Cycle Waste Rec'd	Feed To Evaporator	In Storage End Mos.	* Reserve Cap. In Batches
EAST AREA					
B	751		-169	582	
Evaporator Feed Tank - 106 B	402		+46	448	29
C	427			727	
BX	3175			3175	
BY	2190	43	-414	1819	11
TOTAL	7245	43	-537	6751	40
WEST AREA					
T	1910			1910	
U	382			382	
TX	917	22	3	936	747
Evaporator Feed Tank - 118 TX	90	(708)1	63	735	8
TY	-			-	
TOTAL	3299	730	66	3963	755

* Reserve Capacity at 2800 gallons per batch.
** Not included in totals. **Excludes evaporator feed tanks 106-B and 118-TX.

NOTES: (1) Adjusted to include 712,000 gallons of bottoms supernate received from 106-TX bottoms tank to 118-TX, and loss of 4,000 gallon heel in 114-TX tank during waste storage.

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WASTE STORAGE STATUS

As of 9-30-52

SECOND CYCLE WASTES - SECTION 5 WASTES

	Cascade In Use	Gallons x 10^3		Estimated Reserve Capacity In Batches
		Cribbed	Sludge	
EAST AREA	110-111-112-B	89	414	3020**
WEST AREA	110-111-112-T	85	717	2352
TOTAL		174	1131	5372

REMARKS: 61,000 gallons to Sec. 5 - 200-E, and 68,000 to Sect. 5- 200 W, included in crib figures.

**Reserve calculations based on storage of combined 5-6 & 2nd cycle only at 275 gallons of sludge per run.

**Reserve calculations based on storage of combined 5-6, 2nd cycle, and 224 wastes at 350 gallons of sludge per run.

224 BUILDING WASTES

	Cascade In Use	Gallons x 10^3		Estimated Reserve Capacity In Batches
		Cribbed	Sludge	
EAST AREA	204-203-202-B	42	170	476
WEST AREA	204-201-T 110-111-112-T	Inactive cascade 60	187 Sludge of reserve shown with 2nd cycle.	-
TOTAL		102	357	476

REMARKS:

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WASTE [redacted] STATUS

[redacted]

27839

As of 9-30-52

TBP WASTE

FARM	Gallons x 10^3			
	In Storage Beg Mo	TBP Waste Received	In Storage End Mo	Reserve Capacity
EAST AREA				
B				
C				2,443
BX				
BY				414 (1)
TOTAL		None		2,857
WEST AREA				
T				1,263
U				216
TX	380	---	380	
TY				3,110
TOTAL	380	None	380 (4)	4,589

Supernate tanks, 109 BY and 115 TX not included in the above.

Gallons x 10^3 of condensate to crib 840 (2) (3).

Ratio* Feed to Waste

During Month ---

Cumulative to date ---

* Feed adjusted to metal waste equiva[redacted]

REMARKS: 103-C to pump overground to 109-C (1st cycle tank) instead of 109-BY during October to permit slurrying of 101-102-103-C cascade.

(1) Space released in 107-BY by first cycle evaporation.

(2) Includes 0.48 tons at decontaminated and depleted uranium.

(3) In addition to the amount cribbed, 336,000 gallons containing 2.97 tons of decontaminated and depleted uranium were ditched.

(4) 371,000 gallons of this was [redacted]

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WASTE STORAGE STATUS

[redacted] As of 7-31-52

[redacted]

WASTE EVAPORATOR

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FARM	Gallons x 10^3			
	Feed	Bottoms	Condensate To Cribs	Reserve Bottoms Space
EAST AREA				
U	123	162	375	554
C				
BX				
BY	414			
TOTAL	537	162	375	[illegible] (1)
WEST AREA				
T				
U				
TY	66	30	36	720
TX				
TOTAL	66	30	36	720 (2)

REMARKS: (1) Bottoms reserve has increased due to pumping 105-B and part of 104-B to evaporator feed tank. (2) Temporary increase due to contents of 116 TX bottoms tank being pumped to [illegible] feed tank for re-evaporation.

REDOX WASTE

FARM	Gallons x 10^3			Reserve Capacity In Batches
	In Storage Beg. Mo.	Waste Received	In Storage End Mo.	
S Farm	2,038	391	2,429	[illegible] (1)
FU Farm				
TOTAL	2,038	391	2,429	[illegible]

Gallons x 10^3 of condensate to crib 342

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* Will be considered as Redox reserve capacity when empty.

** Reserve Capacity at [redacted] gallons per U-3 equivalent batch.

REMARKS: (1) Additional Redox reserve space is now available in tanks 241-U-110-111-112 [redacted] and 241-TY-101-[redacted] 1,000 and 1,505,000 gal. respectively.
(2) Based on average [redacted]

[redacted]

[redacted]

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STATUS OF WASTE FARMS

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	Type of Waste	Gallons x 10^3 Capacity	Gallons x 10^3 Stored	Reserve Cap. Gallons x 10^3	Reserve Cap. In Batches	Remarks
B Farm						
101	MW	530	530			
102	MW	530	530			
103	MW	530	519			
104	1C	530	410	120	EB	Partially pumped to 106-B. Not down to sludge.
105	EB	530	172	358		Completed pumping - [illegible]/52.
106	1C	530	445	82	"	Drain. Recc Tank.
107	EB	530	461	69	EB	Active bottom tk.-Sect. Area evap.
108	EB	530	530			
109	EB	530	528	7	EB	
110	2C	530	530			Active cascade-B plant. 2nd cycle &
111	2C	530	530			Section 5.
112	2C	542	542			Tank 112-B. Vacuum to crit.
201	224	54.5	54.5			Active cascade - [illegible]
202	224	54.5	54.5			
203	224	54.5	54.5			
204	224	54.5	54.5			
C Farm						
101	MW	530	530			
102	MW	530	530			
103	MW	530	519			Supernate to pump to [illegible]
104	MW	530	530			
105	MW	530	530			
106	MW	530	519			
107	1C	530	399	131	TBP	
108	1C	530	34	496	TBP	
109	1C	530	10	515	TBP	To receive TBP supernate from 103-C.
110	1C	530	231	299	TBP	
111	1C	530	36	494	TBP	
112	1C	530	17	508	TBP	
201	MW	54.5	54.5			
202	MW	54.5	54.5			
203	MW	54.5	54.5			
204	MW	54.5	54.5			

MW = Metal Waste
1C - First Cycle Waste
2C - Second Cycle Waste
R - Redox Waste

224 - 224 Filtge Waste
TBP - TBP Waste
EB - Evaporator Bottoms

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[redacted]

[redacted]

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STATUS OF WASTE FARMS

[redacted]

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	Type of Waste	Gallons x 10^3 Capacity	Gallons x 10^3 Stored	Reserve Capacity Gallons x 10^3	Reserve Capacity In Batches	Remarks
BX Farm						
101	MW	530	530			
102	MW	530	467	63	23	
103	MW	530	530			
104	MW	530	530			
105	MW	530	530			
106	MW	530	530			
107	1C	530	530			
108	1C	530	530			
109	1C	530	530			
110	1C	530	530			
111	1C	530	525			
112	1C	530	530			
BY Farm						
101	MW	758	758			
102	MW	758	758			
103	MW	758	664	83	31	
104	MW	758	758			
105	MW	758	491	267	99	
106	MW	758	---	747	277	
107	1C	758	344	414	TBP	Partially pumped 9-25 to 9-28.
108	1C	758	753			
109	TBP	758	---	758		TBP supernate tank-200 East Area.
110	1C	758	722	31	11	Active-1st cycle tank-221-B.
111	MW	758	758			
112	MW	758	294	453	168	Active - MW tank - 221-B.
T Farm						
101	MW	530	530			
102	MW	530	530			
103	MW	530	519			
104	1C	530	530			Contents of cascade will be aged 1 year - 3/1/53, after which evaporation may be started.
105	1C	530	530			
106	1C	530	528			
107	1C	530	245	285	TBP	
108	1C	530	73	457	TBP	
109	1C	530	4	515	TBP	

* MW - Metal Waste
1C - First Cycle Waste
2C - Second Cycle Waste
R - Redox Waste

[redacted] Bridge Waste
[redacted]

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STATUS OF WASTE TANKS

[redacted]

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	* Type of Waste	Gallons x 10^3 Capacity	Gallons x 10^3 Stored	Reserve Capacity Gallons x 10^3	Reserve Capacity In Batches	Remarks
T Farm						
110	224 & 2C	530	530			Active cascade - T plant - 2nd
111	224 & 2C	530	530			cycle, 224 and Sect. 5 Wastes.
112	224 & 2C	569	569			112-T cascades to crib.
201	224	54.5	54.5			
202	224	54.5	54.5			
203	224	54.5	54.5			
204	224	54.5	54.5			
U Farm						
101	MW	530	530			Cascade now slurrying for feed to TBP Plant.
102	MW	530	530			
103	MW	530	519			
104	MW	530	530			
105	MW	530	530			
106	MW	530	519			
107	MW	530	530			
108	MW	530	530			
109	MW	530	519			
110	1C	530	336	194	71 **	Cascade being held as Redox
111	1C	530	14	516	187	waste reserve.
112	1C	530	32	498	181	
201	TBP	54.5	---	52.5	TBP	
202	TBP	54.5	---	54.5	TBP	
203	TBP	54.5	---	54.5	TBP	
204	TBP	54.5	---	54.5	TBP	
TX Farm						
101	MW	758	758			
102	MW	758	758	8		
103	MW	758	758			
104	MW	758	750			
105	MW	758	758			
106	MW	758	95	663	246	Pumped to 1-4 TX on [illegible] over-
107	MW	758	758			ground.
108	MW	758	718	29	11	Active tank - 221-T - metal waste.
109	1C	758	758	---	---	Active cascade - 221-T - first
110	1C	758	178	580	207	cycle.
111	1C	758	---	758	271	
112	1C	758	---	753	269	

* MW - Metal Waste
* 1C - First Cycle Wa[redacted]
2C - Second Cycle
R - Redox Waste
H-2902.10-X 4-52 [redacted]

224 - 224 Bldg. Waste
[redacted]P - TBP Waste
[redacted] - Evaporator Bottoms

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STATUS OF WASTE FARMS

[redacted]

	* Type of Waste	Gallons x 10^3 Capacity	Gallons x 10^3 Stored	Reserve Capacity Gallons x 10^3	Reserve Capacity In Batches	Remarks
TX Farm (cont'd)						
113	EB	758	750	8	---	
114	MW	758	666	81	30	Received from 106-TX, 9/6/52.
115	TBP	758	758	---	---	TBP supernate tank-200West Area
116	EB	758	46	712		Pumped to 118 TX for starting re-evaporation.
117	EB	758	756	---	---	
118	1C	758	735	23	8	Evaporator feed tank.
TY Farm						
101	R	758		758	276**	
102	R	758		747	272	
103	TBP	758		758	276	
104	TBP	758		747	272	
105	TBP	758		758	276	
106	TBP	758		747	272	
S Farm						
101	R	758		758	276	
102	R	758		758	276	
103	R	758		747	272	
104	R	758		758	276	
105	R	758		758	276	
106	R	758		747	272	
107	R	758	458	300	109	Active tank - Redox wastes.
108	R	758	---	758	276	
109	R	758	---	747	272	
110	R	758	774	---	---	
111	R	758	755	---	---	
112	R	758	442	305	111	

* MW - Metal Waste
1C - First Cycle Waste
2C - Second Cycle Waste
R - Redox Waste

224 - 224 Bldg. Waste
TBP - TBP Waste
EB - Evaporator Bottoms

** - Reserve figures for storage in tank farm 241-S and tanks 241-U-110-112; 241-TY-101-102. Based on average plant performance of 2,750 gallons per batch.

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WASTE STATUS SUMMARY Separations Section [redacted]

Period: OCTOBER, NOVEMBER AND DECEMBER, 1952

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WASTE STATUS SUMMARY

Month of October 1952

	B Plant	T Plant
Runs of Metal Waste to waste farm during the month	2 Bicarb. flushes	
Gals. increase in Metal Waste volume	2,000	58,9[illegible]
Gals./Run average Metal Waste volume	- (1)	2,[illegible]
Reserve Gals. available for Metal Waste on 11-1-52	1,556,000 (2)	708,[illegible]
Runs of First Cycle Waste sent to farm during the month	2 AW	
Gals. increase in First Cycle Waste volume	9,600	55,[illegible]
Gals./Run average First Cycle Waste volume	- (1)	2,[illegible]
Reserve Gals. available for First Cycle Waste on 11-1-52	304,000 (2)	2,076,[illegible]

Remarks: (1) Since B-Plant batches have been of abnormal sizes, no averages have been calculated. (2) In November 5-6, 2nd cycle, 1st cycle, and stack drainage will be cribbed after passing through 110-111-112 B cascade. Metal waste flushes will be cribbed after passing through 200-BP tank cascade. Because of these changes, B-Plant no longer requires first cycle and metal waste reserves. (3) Underground line connecting 105 and 105-TX (T-Plant active metal waste cascade) was plugged at month's end. This has resulted in an unusually high liquid level in the 105-TX tank. Metal waste averages shown above are based on the assumption that none of this material has been lost to the ground.

	S Plant
Bismuth Phosphate Batch Equivalent	121.[illegible]
Gallons increase in fission product waste storage	323,0[illegible]
Tons of Uranium, (as charged), processed to contribute to make-up of waste	73.[illegible]
Gallons/batch equivalent to 6-3 MS	2,65[illegible]
Gallons per ton of Uranium processed (1)	4,3[illegible]
Reserve Gallons available for Redox waste on 10-31-52 (2)	6,319,0[illegible]

Remarks: (1) This average shows an 8.5% increase per ton over September. This was due to increased waste volumes resulting from equipment changes in the third uranium cycle. The theoretical effect of these changes is 8.7%. (2) These reserves are those in tank farm 241-S only.

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WASTE STORAGE STATUS

As of 10-31-52

METAL WASTE

FARM	Gallons x 10^3 In Storage Beg. Mo.	Metal Waste Received	In Storage End Mo.	Reserve Capacity In Batches
EAST AREA				
B	1,579		1,579	
C	3,374		3,374 (1)	
BX	3,117		3,117	23
BY	4,481	57 (2)	4,538	553
TOTAL	12,551	57	12,608	576
WEST AREA				
T	1,579		1,579	
U	4,737		4,737 (1)	
TX	6,019	59	6,078	265
TY	---		---	---
TOTAL	12,335	59	12,394	265

REMARKS: *Reserve capacity at 2,700 Gallons per batch.

(1) Cascades 101-103 C and 101-103 U slurrying for T3P feed.

(2) Includes 2,000 gallons from B-Plant and 55,000 gallons from T3P-Plant.

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WASTE STORAGE STATUS

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of 10-31-52

URANIUM CONTENT
METAL WASTE

FARM	TONS OF DEPLETED URANIUM			
	In Storage Beg. Mo.	Received	* To TBP	In Storage End Mo.
EAST AREA				
B	383			383
C	837	1.32 (1)	392 (4)	446.32
BX	758			758
BY	1,057.76	0.61 (2)		1,058.37
TOTALS	3,035.76	1.93		2,645.69
WEST AREA				
T	374			374
U	770	0.45 (3)		770.45
TX	1,381.63	13.62		1,395.25
TY				
TOTAL	2,525.63	14.07		2,539.70 (5)

* Tons of uranium transferred as metal waste to the Waste Removal Group for slurrying.

REMARKS: (1) Uranium in recovered nitric acid sent to C Farm in September and October.

(2) 55,000 gallons from TBP-Plant to 111-112 BY.

(3) Uranium in recovered nitric acid sent to U-Farm in September and October.

(4) 392 tons of uranium in cascade 101-103 C transferred to TBP account.

(5) 405 tons of uranium in cascade 101-103 U not included in this total.

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WASTE STORAGE STA[illegible]

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As of 10-31-52

FIRST CYCLE WASTE AND COATING WASTES

T AND B PLANTS

FARM	Gallons x 10^3				
	In Storage Beg. Mo.	1st Cycle Waste Rec'd	Feed To Evaporator	In Storage End Mo.	Reserve In Batch
EAST AREA					
B	582			582	
Evaporator Feed Tank - 106 B	448		201	247	101
C	727			727	
BX	3,175			3,175	
BY	1,819	10		1,829	8
TOTAL	6,751	10	201	6,560	109
WEST AREA					
T	1,910			1,910	
U	382			382	
TX	936	55		991	727
Evaporator Feed Tank - 118 TX	735		17	718	14
TY	---				
TOTAL	3,963	55	17	4,001	741

Reserve Capacity at 2,800 gallons per batch

( ) Not included in totals.

REMARKS:

[redacted]

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WASTE STORAGE STATUS

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As of 10-31-52

SECOND CYCLE WASTES

	Cascade In Use	Gallons x 10^3 Cribbed	Gallons x 10^3 Sludge	Estimated Reserve Capacity In Batches
EAST AREA	110-111-112-B	82	444	3,020 **
WEST AREA	110-111-112-T	162	725	2,344 ***
TOTAL		244	1,139	5,364

REMARKS: 59,000 gallons from Section 5, 221-B and 114,000 gallons from Section 5, 221-T included in cribbed figures.

** Reserve calculations based on storage of combined 5-6 and second cycle only at 275 gallons of sludge per run.

*** Reserve calculations based on storage of combined 5-6, second cycle and 224-T wastes at 350 gallons of sludge per run.

224 BUILDING WASTES

	Cascade In Use	Gallons x 10^3 Cribbed	Gallons x 10^3 Sludge	Estimated Reserve Capacity In Batches
EAST AREA	204-201-202-B	12	176	476
WEST AREA	204-201-T 110-111-112-T	Inactive Cascade 96	187 Included with 2nd cycle figures	---
TOTAL		146	357	476

REMARKS:

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WASTE STORAGE STATUS

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As of 10-31-52

TBP WASTE

Date	Gallons x 10^3			
	In Storage Beg Mon	TBP Waste Received	In Storage End Mon	Waste Calcined
Last Mon				
[illegible]				2,443
[illegible]				
[illegible]		(55) (1)		414
TOTAL				2,857
This Mon				
[illegible]				1,263
[illegible]				216
TX	380		380	
[illegible]				3,110
TOTAL	380		380	4,589

Supernate tanks, 109 BY and 115 TX not included in the above.

Gallons x 10^3 of condensate to crib 854 (2) (3)

Ratio - Feed to Waste

During Month ---

Cumulative to date ---

Feed adjusted to metal waste equivalent

Remarks: (1) Stored in unprocessed metal waste tanks 111-112 BY; this material will be re-worked. Shown as metal waste figures, includes 0.61 tons of depleted uranium.

(2) Includes amount cribbed to 216-W and 216-W crib - contained 0.84 tons of depleted uranium.

(3) In addition to this figure, 405,000 gallons of solution containing 2.40 tons of depleted uranium were ditched.

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WASTE EVAPORATOR

Date	Gallons x 10^3			
	Feed	Bottoms	Condensate To Crib	Storage Bottom Space
Start Size				
2	201	64	137	490
2				
28				
29				
TOTAL	201	64	137	490
Start Size				
2				
2				
28	17	46 (1)	-29 (1)	674
29				
TOTAL	17	46	-29	674

(1) Bottoms figures exceeded feed because of evaporator flushes, hence recovered space or condensate is negative.

REDOX WASTE

Date	Gallons x 10^3			
	In Storage Beg. Yrs.	Waste Received	In Storage End Yrs.	Storage Capacity In Gallons
2 Years	2,429	323	2,752	2,300
29 Years				
TOTAL	2,429	323	2,752	

Gallons x 10^3 of condensate to crib 300

Gallons to consider as redox storage capacity

Average Capacity of 2,750 gallons per 6-9 equivalent batch

REMARKS: A total of 2,713,000 gallons of storage space immediately available 241-TX and 241-AY TANKS

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[redacted] HW-27[illegible]

[redacted] SEC[redacted]

STATUS OF WASTE TANKS

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	* Type of Waste	Gallons x 10^3 Capacity	Gallons x 10^3 Stored	Reserve Cap. Gallons x 10^3	Reserve Cap. In Batches	Remarks
B Farm						
101	MW	530	530			
102	MW	530	530			
103	MW	530	519			
104	1C	530	410	120	EB	
105	EB	530	172	358	EB	
106	1C	530	247	283	TBP	Evaporator feed tank
107	EB	530	525	5	EB	Active bottoms tank-East Area [illegible]
108	EB	530	530	—	—	
109	EB	530	518	7	EB	
110	2C	530	530			2nd cycle and 5-6 waste, B Flam[illegible] active cascade
111	2C	530	530			
112	2C	542	542			112-B cascades to crib
201	224	54.5	54.5			Active cascade, 224-B
202	224	54.5	54.5			
203	224	54.5	54.5			
204	224	54.5	54.5			
C Farm						
101	[illegible]	530	530			Cascade now slurrying for TBP [illegible]
102	[illegible]	530	[illegible]			
103	[illegible]	530	519			
104	[illegible]	530	530			
105	[illegible]	530	530			
106	[illegible]	530	519			
107	1[illegible]	530	399	131	TBP	
108	1[illegible]	530	[illegible]	[illegible]	TBP	
109	1[illegible]	530	10	515	TBP	Temporary supernate tank for [illegible]-Farm operations
110	1[illegible]	530	231	299	TBP	Active cascade TBP operations
111	[illegible]	530	36	494	[illegible]	
112	[illegible]	530	17	508	TBP	
201	MW	54.5	52.5			
202	MW	54.5	54.5			
203	MW	54.5	54.5			
204	MW	54.5	54.5			

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* MW - Metal Waste
1C - First Cycle Waste
2C - Second Cycle Waste
R - Redox Waste

224 - 224 Bldg. Waste
TBP - TBP Waste
EB - Evaporator Bottoms

[redacted]

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[redacted] SECURITY INFORMATION

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HW-27[illegible]

RESTRICTED SECURITY INFORMATION

DECLASSIFIED

	Type of Waste	Gallons x 10^3 Capacity	Gallons x 10^3 Stored	Reserve Capacity Gallons x 10^3	Reserve Capacity In Batches	Remarks
BX Farm						
101	MW	530	530			
102	MW	530	467	63	23	
103	MW	530	530			
104	MW	530	530			
105	MW	530	530			
106	MW	530	530			
107	1C	530	530			
108	1C	530	530			
109	1C	530	530			
110	1C	530	530			
111	1C	530	525			
112	1C	530	530			
BY Farm						
101	MW	758	758			
102	MW	758	758			
103	MW	758	694	63	31	
104	MW	758	758			
105	MW	758	491	267	99	
106	MW	758	---	247	277	
107	1C	758	[illegible]	[illegible]	[illegible]	Pumping to 100-B not complete
108	1C	758	[illegible]			
109	TBP	758	---	758		TBP supernate tank - 200C E Area
110	1C	758	732	21	8	Abandoned as 1st cycle cascade
111	MW	758	[illegible]	---	---	
112	MW	758	[illegible]	[illegible]	[illegible]	Abandoned by permit as metal waste cascade. Waste will be cribbed.
T Farm						
101	MW	530	530			
102	MW	530	530			
103	MW	530	519			
104	1C	530	530			1 year decay complete 3/31/53
105	1C	530	530			Evaporation scheduled to
106	1C	530	528			start 4/1/53
107	1C	530	245	285	TBP	
108	1C	530	73	457	TBP	
109	1C	530	4	515	TBP	

- MW - Metal Waste
- 1C - First Cycle Waste
- 2C - Second Cycle Waste
- R - Redox Waste
- 224 - 224 Bldg. Waste
- TBP - TBP Waste
- EB - Evaporator Bottoms

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RESTRICTED

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SECURITY INFORMATION

[redacted]

HN-278[illegible]

[redacted] SECURITY INFORMATION

STATUS OF WASTE TANKS

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	Type of Waste	Gallons x 10^3 Capacity	Gallons x 10^3 Stored	Reserve Capacity Gallons x 10^3	Reserve Capacity In Batches	Remarks
T Farm						
110	224 & 2C	530	530			Active cascade-T Plant; 2nd
111	224 & 2C	530	530			cycle, 224 & Sect. 5 wastes
112	224 & 2C	569	569			Tank 112-T cascades to crib
201	224	54.5	54.5			
202	224	54.5	54.5			
203	224	54.5	54.5			
204	224	54.5	54.5			
U Farm						
101	MW	530	530			Cascade now slurrying for fe
102	MW	530	530			to TBP
103	MW	530	519			
104	MW	530	530			
105	MW	530	530			
106	MW	530	519			
107	MW	530	530			
108	MW	530	530			
109	MW	530	519			
110	1C	530	[illegible]	[illegible]	[illegible]	Cascade being held as Redox
111	1C	530	14	516	[illegible]	waste reserve
112	1C	530	[illegible]	495	[illegible]	
201	TBP	54.5		[illegible]	TBP	
202	TBP	54.5		[illegible]	TBP	
203	TBP	54.5		[illegible]	TBP	
204	TBP	54.5		54.5	TBP	
TX Farm						
101	[illegible]	758	[illegible]			
102	[illegible]	758	[illegible]			
103	[illegible]	758	[illegible]			
104	[illegible]	758	[illegible]			
105	MW	758	[illegible]	[illegible]	[illegible]	Overflow to 106-TX plugged a
106	MW	758	102	[illegible]	[illegible]	month end
107	MW	758	[illegible]			Supernate temp. [illegible]° F,
108	MW	758	732			10-31-52
109	1C	758	758	---	---	Active cascade, 221-T, 1st
110	1C	758	233	525	[illegible]	
111	1C	758	---	758	221	
112	1C	758	---	753	[illegible]	

* MW - Metal Waste
* 1C - First Cycle Waste
2C - Second Cycle Waste
R - Redox Waste

224 - 224 Bldg. Waste
TBP - TBP Waste

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[redacted]

[redacted] SECURITY INFORMATION

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SECURITY INFORMATION

HW-27

STATUS OF WASTE FARMS

DECLASSIFIED

	* Type of Waste	Gallons x 10^3 Capacity	Gallons x 10^3 Stored	Reserve Capacity Gallons x 10^3	Reserve Capacity In Batches	Remarks
TX Farm (cont'd)						
113	EB	758	750	8	EB	
114	MW	758	666	81	30	
115	TBP	758	758			TBP supernate tank, 2-West Area
116	EB	758	92	666	EB	Active bottoms tank, 2-West Area
117	EB	758	756			
118	1C	758	718	40	14	Evap. feed tank, 2-West Area
TY Farm						
101	R	758	---	758	278 **	
102	R	758	---	747	272	
103	TBP	758	---	758	---	
104	TBP	758	---	747	---	
105	TBP	758	---	758	---	
106	TBP	758	---	747	---	
S Farm						
101	R	758	---	758	278 **	
102	R	758	---	756	276	
103	R	758	---	747	272	
104	R	758	---	758	276	
105	R	758	---	756	276	
106	R	758	---	747	272	
107	R	758	763	---	---	Filled 10-30-52
108	R	758	17	741	[illegible]	Active tank, Redox waste
109	R	758	---	747	[illegible]	
110	R	758	774	---	---	
111	R	758	735			
112	R	758	442	305	[illegible]	

* MW - Metal Waste
1C - First Cycle Waste
2C - Second Cycle Waste
R - Redox Waste

224 - 224 Bldg. Waste
TBP - TBP Waste
EB - Evaporator Bottoms

** - Reserve figures for storage in tank farm 241-S-101-109 and tanks 241-S-110-112; 241-TY-101-102 based on average plant performance of 2,750 gallons per batch.

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SECURITY INFORMATION

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[redacted]

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WASTE STATUS SUMMARY

(13)

[redacted]

Month of November 1952

	In Plant	2 Plant
Tons of Total Waste to waste farm during the month	[redacted]	[redacted]
Gallons Increase in Total Waste volume	[redacted]	82,230
Gallons/Run average Total Waste volume	[redacted]	[redacted]
Reserve Gallons available for Total Waste on 11-30-52	1,556,000 (1)	652,000
Tons of First Cycle Waste sent to farm during the month	[redacted]	[redacted]
Gallons Increase in First Cycle Waste volume	[redacted]	63,250
Gallons/Run average First Cycle Waste volume	[redacted]	2,530
Reserve Gallons available for First Cycle Waste on 11-30-52	386,000 (1)	2,013,000

Remarks (1) Reserve space no longer required for Replant since all wastes from buildings are being cribbed.

[redacted]

	2 Plant
Bismuth Phosphate Batch Equivalent	182.73
Gallons Increase in Bismuth product waste storage	473,000
Tons of Uranium (as charged) processed to certain butt of make-up of waste	114.10
Gallons/batch equivalent to 6-3 TS	2,589
Gallons per ton of Uranium processed	4,145
Reserve Gallons available for Redox waste on 11-30-52	(1) 5,846,000

Remarks (1) These reserves are those in tank farm 241-S only.

[redacted]

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[redacted]

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[redacted]

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[redacted]

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14

Waste Storage Status

[redacted] as of 11-30-52

TOTAL WASTE

[redacted]

FARM	In Storage Gallons	Total Waste Received	In Storage Gallons	Empty Capacity in Gallons
EAST AREA				
B	1,579		1,579	
[illegible]	3,374		3,374 (1)	
BX	3,117		3,117	23
BY	4,538	---	4,538	553
TOTAL	12,608	---	12,608	576
WEST AREA				
T	1,579		1,579	
U	4,737		4,737 (1)	
TX	6,078	82	6,160	241
TY	---		---	
TOTAL	12,394	82	12,476	241

REMARKS: *Reserve capacity of 2,700 Gallons [illegible]batche

(1) Cascades 101-103 C and 101-103 U are now serving as feed tanks to TBP Plant. These cascades will be indicated as full until the entire cascade has been released.

[redacted]

[redacted]

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[redacted]

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(15)

[redacted]

No. 11-30-52

[redacted]

[redacted]

Item	In Storage Beg. Bal.	Received	To TBP	In Storage End Bal.
[redacted]				
[redacted]	383			383
[redacted]	446.32 (2)	3.40 (1)		449.72
[redacted]	758			758
[redacted]	1,058.37			1,058.37
TOTAL	2,645.09	3.40		2649.09
[redacted]				
[redacted]	374			374
[redacted]	770.45 (2)	2.90 (1)		773.35
[redacted]	1,395.25	16.27		1,411.52
[redacted]	---			---
TOTAL	2,539.70	19.17		2558.87

* Tons of uranium transferred as metal equal to the uranium removed from [illegible] for slurrying [illegible]

(1) 6.30 tons [illegible] addition to [illegible] at blend tanks in form of recovered nitric acid. This uranium will be fed back through the TBP Plant.

(2) 392 tons in cascade 101-103 C and [illegible] tons in cascade 101-103 U are now being fed to TBP Plant. These quantities are not included in the above figures.

[redacted]

[redacted]

[redacted]

[redacted]

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HW-27840

[redacted]

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[redacted]

11-30-

[redacted] AND COATING WASTES
J AND B PLANTS

[redacted]

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]					
B	582			582	
Evaporator Feed Tank to 206 B	247		82	165	130
[redacted]	727			727	
BX	3,175			3,175	
BP	1,829	---		1,829	8
TOTAL	6,560	---	82	6,478	138
[redacted]					
[redacted]	1,910			1,910	
[redacted]	362			362	
TX	991	63		1,054	705
Evaporator Feed Tank to [redacted]	718		---	718	14
[redacted]	---			---	
TOTAL	4,001	63	---	4,064	719

[redacted] 2,800 [redacted]

[redacted]

[redacted]

[redacted]

[redacted]

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HN - 27840

[redacted]

WASTE STORAGE STATUS

DECLASSIFIED

[redacted] 11-30-52

SECOND CYCLE WASTE

[redacted]

	Cascade No.	Gallons		Estimated Reserve Capacity in Gallons
		Crudes	Sludge	
EAST AREA	110-111-112 B	131 (1)	414	3,020
WEST AREA	110-111-112 T	178	733	2,321
TOTAL		309	1,147	5,341

REMARKS (1) Includes metal waste, first cycle, section five, and second cycle wastes resulting from cleanouts.

** Reserve calculations based on storage of combined 5s6 and second cycle only at 275 gallons of sludge per run.

*** Reserve calculations based on storage of combined 5s6, 2nd cycle and 224-T waste at 350 gallons of sludge per run.

224 ROTORS WASTE

	Cascade No.	Gallons		Estimated Reserve Capacity in Gallons
		Crudes	Sludge	
EAST AREA	204-203-202 B	24	170	476
WEST AREA	204-201-T 110-111-112 T	79 Inactive cascade	187 Incl. with 2nd cycle figures	---
TOTAL		103	357	476

REMARKS

[redacted]

[redacted]

[redacted]

[redacted]

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[redacted]

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WASTE STORAGE STATUS

As of 11-30-52

[redacted]

FARM	Gallons x 10^3 In Storage Beg. Mo.	TBP WASTE	In Storage	Reserve Capacity
EAST AREA				
B				
C		362	362	1,580 (3)
BX				
BY	55 (1)		55	414
TOTAL	55	362	417	1,994
WEST AREA				
T		426	426	831
U				
TX	380 (2)		380	
TY				3,010
TOTAL	380	426	806	3,841

Supernate tanks, 109 BY and 115 TX not included in the above

Gallons x 10^3 of condensate to crib 2,055

Ratio: ~~Feed-to-waste~~ WASTE TO FEED

During Month 0.95

Cumulative to date 1.65 (4)

* Feed adjusted to metal waste equivalent

REMARKS: (1) High waste in metal waste cascade 111-112 BY is reworkable.

(2) High waste [redacted]

(3) Tank 109 C not included as reserve space since this space is not immediately available.

(4) Includes high waste [redacted]

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[redacted]

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[redacted]

HW-27848

WASTE STORAGE STATUS

[redacted]

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As of 11-30-52

(19)

WASTE EVAPORATOR

FARM	Gallons x 10^3 Feed	Bottoms	Condensate To Cribs	Reserve Bottoms Space
EAST AREA		[redacted]		
B	82.5	20	62.5	489
C				
BX				
BY				
TOTAL	82.5	20	62.5	489
WEST AREA				
T				
U				
TX	0	0	0	676 (1)
TY				
TOTAL	0	0	0	676 (1)

REMARKS: (1) 2,000 gallon reserve increase in 113 TX because of volume contraction of liquid following cooling below boiling point toward atmospheric temperature.

REDOX WASTE

FARM	Gallons x 10^3 In Storage Beg. Mo.	Waste Received	In Storage End Mon.	** Reserve Capacity In Batches
S Farm	2,751	473	3,224	2,245
SY Farm				
TOTAL	2,751	[redacted]		2,245

Gallons x 10^3 of Condensate to cribs 406

[redacted] to consider as Redox waste capacity when empty

[redacted] Capacity of 2,600 gallons [redacted] batch

REMARKS: A total of 2,713,000 gallons of storage space is available in 241-TY and 241-U tank farms.

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STATUS OF WASTE TANKS

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[redacted] DECLASSIFIED W-2784

Farm		* Type of Waste	Gallons x 10^3 Capacity	Storage	Reserve Cap.		Remarks
101		MW	530	530			
102		MW	530	530			
103		MW	530	519			
104		1C	530	410	120	EB	
105		EB	530	172	358	EB	
106		1C	530	165	365	130	Evaporator feed tank
107		EB	530	531	---		Cascaded to 108-B, 11-25-52
108		EB	530	528	---		Cascaded to 109-B, 11-30-52 *
109		EB	530	514	11	EB	
110	1C -	2C-5-6	530	530			Active cascade, B Plant receives
111	1C -	2C-5-6	530	530			flushes of Sec. 5, 1st cycle &
112	1C -	2C-5-6	542	542			2nd cycle lines
201	MW	224	54.5	54.5			Active cascade, B Plant receives
202	MW	224	54.5	54.5			flushes of 224-B Bldg. & MW lines
203	MW	224	54.5	54.5			
204	MW	224	54.5	54.5			
C Farm							
101		MW	530				Cascade now serving as feed to
102		MW	530	779			TBP Plant
103		MW	530				
104		MW	530	530			
105		MW	530	530			
106		MW	530	519			
107		1C	530	399-0	131	TBP	
108		1C	530	31-0	406	TBP	
109		1C-TBP	530	10-486	23	TBP	Temporary supernate tank for
110		1C-TBP	530	231-259	40	TBP	C-Farm MW removal operations
111		1C-TBP	530	36-103	391	TBP	200-E active receiver of TBP
112		1C-TBP	530	17-0	502	TBP	wastes. Overflow 110-C to 111-C
							plugged on 11-15-52
201		MW	54.5	52.5			
202		MW	54.5	54.5			
203		MW	54.5	54.5			
204		MW	54.5	54.5			

[redacted]

* MW - Metal Waste
1C - First Cycle Waste
2C - Second Cycle Waste
R - Redox Waste

[redacted] - 224 Bldg. Waste
TBP - TBP Waste
EB - Evaporator Bottoms

[redacted]

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* Tanks 108 and 109-B were previously in service for storage of evaporator bottoms in May and August respectively. Since those bottoms solutions have cooled, causing a density [redacted]. This has resulted in 19,000 gallons of additional storage space, which was utilized by cascading wastes from 107-B to 108-B to 109-B in November.

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STATUS OF WASTE FARMS

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	Type of Waste	Gallons x 10^3 Capacity	Gallons x 10^3 Stored	Gallons x 10^3	Batches	Remarks
BX Farm						
101	MW	530	530			
102	MW	530	467	63	23	
103	MW	530	530			
104	MW	530	530			
105	MW	530	530			
106	MW	530	530			
107	1C	530	530			
108	1C	530	530			
109	1C	530	530			
110	1C	530	530			
111	1C	530	525			
112	1C	530	530			
BY Farm						
101	MW	758	758			
102	MW	758	758			
103	MW	758	664	83	31	
104	MW	758	758			
105	MW	758	491	267	99	
106	MW	758	---	747	277	
107	1C	758	344	414	TBP	No supernate pumped to 106-B in November
108	1C	758	753			
109	TBP	758	---	758		TBP supernate tank, 200 E area
110	1C	758	732	21	8	
111	MW	758	758	---		
112	MW	758	351	396	147	
BZ Farm						
101	MW	530	530			
102	MW	530	530			
103	MW	530	519			
104	1C	530	530			lower decay complete 3/31/53
105	1C	530	[redacted]			evaporation scheduled to start
106	1C	530	[redacted]			4-1-53
107	1C-TBP	530	245-285	---	TBP	full 11-26-52. Started filling with TBP, 11-17-52, because of
108	1C-TBP	530	73-141	316	TBP	plug in 110-111-112 C cascade
109	1C-TBP	530	[redacted]		TBP	

* MW - Metal Waste
1C - First Cycle Waste
2C - Second Cycle Waste
R - Redox Waste
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224 - 224 Bldg. Waste
TBP - TBP Waste
EB - Evaporator Bottoms

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[redacted]

HW-27840

STATUS OF WASTE FARMS

DECLASSIFIED

	* Type of Waste	Gallons x 10^3 Capacity	Gallons x 10^3 Stored	Reserve Capacity Gallons x 10^3	Reserve Capacity In Batches	Remarks
T Farm						
110	224&2C	530	530	[redacted]		2nd cycle, Sec. 5, and 224 Bldg.
111	224&2C	530	530*			Active cascade, T Plant
112	224&2C	569	569			Tank 112. T cascades to crib
201	224	54.5	54.5			
202	224	54.5	54.5			
203	224	54.5	[illegible]			
204	224	54.5	[illegible]			
U Farm						
101	MW	530				Cascade now serving as feed to TBP Plant
102	MW	530	586			
103	MW	530				
104	MW	530	530			
105	MW	530	530			
106	MW	530	519			
107	MW	530	530			
108	MW	530	530			
109	MW	530	530			
110	1C	530	336	194	75 **	Cascade being held as Redox waste reserve
111	1C	530	14	516	198	
112	1C	530	32	498	192	
201	TBP	54.5		52.5	TBP	
202	TBP	54.5		54.5	TBP	
203	TBP	54.5		54.5	TBP	
204	TBP	54.5		54.5	TBP	
X Farm						
101	MW	758	758			
102	MW	758	758			
103	MW	758	758			
104	MW	758	750			
105	MW	758	766	---	---	Plug removed 11-8-52
106	MW	758	187	571	211	
107	MW	758	766	---	---	
108	MW	758	751	---	---	
109	1C	758	758	[redacted]		Active cascade, 221-T, 1st cycle
110	1C	758	296	[redacted]		
111	1C	758	---	[redacted]	271	
112	1C	758	---	753	269	

* MW - Metal Waste
* 1C - First Cycle Waste
2C - Second Cycle
R - Redox Waste
224 & 224 Bldg. Waste
TBP & TBP Waste
EB - Evaporator Bottoms
** See bottom of page H-2902.11-X

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[redacted]

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[redacted]

HW-27[illegible]

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STATUS OF WASTE FARMS

	Type of Waste	Gallons @ 10^3 Capacity	Stored	Gallons x 10^3	In Batches	Remarks
TX Farm (cont'd)						
113	EB	758	748	10	EB	
114	NW	758	666	[redacted]		
115	TBP	758	7[redacted]			TBP supernate tank, 200-W Area
116	EB	758	92	666	EB	Active bottoms tank, 200-W Evap.
117	EB	758	756			
118	1C	758	718	40	1C	Evap. feed tank, 200-W Area
TY Farm						
101	R	758	---	758	291 **	
102	R	758	---	747	287	
103	TBP	758	---	758	TBP	
104	TBP	758	---	747	TBP	
105	TBP	758	---	758	TBP	
106	TBP	758	---	747	TBP	
[illegible] Farm						
101	R	758	---	758	291 **	
102	R	758	---	758	291	
103	R	758	---	747	287	
104	R	758	---	758	291	
105	R	758	---	758	291	
106	R	758	---	747	287	
107	R	758	763		---	
108	R	758	490	268	103	Active tank, Redox wastes
109	R	758	---	747	287	
110	R	758	774			
111	R	758	755			
112	R	758	442	306	117	

NW = Metal Waste
1C = First Cycle Waste
2C = Second Cycle Waste
R - Redox Waste
22k = 22k Ridge Waste
TBP = TBP Waste
EB = Evaporator Bottoms

** Performance of 2,600 gallons per batch (average of Oct.-Nov. 1952)

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[redacted]

[redacted]

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[redacted]

HW-27847

WASTE STATUS SUMMARY

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(24)

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Month of December, 1952

	B Plant	T Plant
Runs of Metal Waste to waste farm during the month	—	41
Gals. increase in Metal Waste volume	(1)	104,500
Gals./Run average Metal Waste volume	—	2,549
Reserve Gals. available for Metal Waste on 12-31-52	1,545,000 (2)	547,000
Runs of First Cycle Waste sent to farm during the month	—	39
Gals. increase in First Cycle Waste volume	(1)	101,750
Gals./Run average First Cycle Waste volume	—	2,600
Reserve Gals. available for First Cycle Waste on 12-31-52	(3) 136,000	(3) 1,911,000

Remarks: (1) All wastes from B Plant are cribbed after passing through either the 200-B series tanks, or 110-112-B cascade. (See 2nd cycle and 224 wastes) below previous month figure, (2) Decreased 11,000 gallons because of addition of temporary TBP waste to 111-112-BY tanks. (3) Includes empty space in first cycle evaporator feed tanks.

	S Plant
Bismuth Phosphate Batch Equivalent	140.69
Gallons increase in Fission product waste storage	373,000
Tons of Uranium (as charged) processed to contribute to make-up of waste	91.00
Gallons/batch equivalent to 6e3 PS	2,650
Gallons per ton of Uranium processed	(1) 4,096
Reserve Gallons available for Redox waste on 12-31-52	(2) 5,475,000

Remarks: (1) Minor process changes were made which tend toward reducing waste volume per ton of uranium feed by a small percentage. (2) These reserves are those in tank farm 241-S only.

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WASTE STORAGE STATUS

As of 12-31-52

METAL WASTE

PLANT	Gallons x 10^3 In Storage Beg. Mo.	Metal Waste Received	In Storage End Mo.	Reserve Capacity In Batches
EAST AREA				
B	1,579		1,579	
C	3,374 (2)		3,374	
BX	3,317		3,317	23
BY	4,538	11 (2)	4,549	550
TOTAL	12,608	11	12,619	573
WEST AREA				
T	1,579		1,579	
U	4,737 (1)		4,737	
TX	6,160	105	6,265	203
TY				
TOTAL	12,476	105	12,581	203

REMARKS: Reserve capacity of 2,700 Gallons or better

(1) Cascades 101-103-C and 101-103-U are now serving as TBP tanks to TBP Plant. These cascades will be indicated as full until the entire cascade has been released.

(2) High waste from TBP Plant return to metal waste storage.

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WASTE STORAGE STATUS

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URANIUM CONTENT
METAL WASTE

FARM	TONS OF DEPLETED URANIUM			
	In Storage Dec. 31st	Blended	* To TBP	In Storage End No.
EAST AREA				
B	383			383
C	(2) 449.72	1.35 (3)		451.07
BY	758			758
BY	1,058.37	0.67 (1)		1,059.04
TOTAL	2,649.09	2.02		2,651.11
WEST AREA				
T	374			374
U	(2) 772.35	1.37 (3)		774.72
TY	1,411.52	24.58		1,436.10
TY				
TOTAL	2,558.87	25.95		2,584.82

* Tons of uranium transferred as metal waste to the Waste Removal Group for stripping.

REMARKS: (1) 0.67 tons of high TBP waste sent to cascade 111-112-BY

(2) 392 tons in cascade 101-103-C and 405 tons in cascade 101-103-U are now being fed to TBP Plant. These quantities are not included in the above total.

(3) 2.72 tons of uranium returned to tank farm blend tanks in form of recovered nitric acid.

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WASTE STORAGE STATUS

As of 12-31-52

FIRST CYCLE WASTE AND COATING WASTE

T AND B PLANTS

FARM	Gallons x 10^3				
	In Storage Beg. No.	1st Cycle Waste Rec'd	Feed To Evaporator	In Storage End No.	Reserve Cap. In Batches*
EAST AREA					
B (1)	2,155	(197)		2,352	
Evaporator Feed Tank - 106 B	165		(250)	415	41
C (2)	727			727	
BX	3,175		(23) (3) 579	2,573	
BY	1,829		343	1,486	8
TOTAL	8,051	(197)	672(423)	7,553	49
WEST AREA					
T	1,910			1,910	
U (2)	382			382	
TX	3,316	102		3,418	669
Evaporator (1) Feed Tank @ 118 TX	718			718	14
TY					
TOTAL	6,326	102		6,428	683

* Reserve Capacity of 2,800 gallons per batch

( ) Not included in totals

REMARKS: All figures have been revised to include first cycle wastes in storage as evaporator bottoms. (1) Figures represent bottoms storage only. (2) First cycle waste as non-evaporable sludge only. (3) 23,000 gallons of first cycle waste lost to ground during pumping of 108 BX supernatant liquid to 106-B evaporator feed tank on 12-16-52.

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WASTE STORAGE STATUS

As of 12-31-52

SECOND CYCLE WASTES

	Cascade In Use	Gallons x 10^3		Estimated Reserve Capacity In Batches
		Cribbed	Sludge	
EAST AREA	110-111-112-T	97 (1)	414	3,020
WEST AREA	110-111-112-T	218	734	2,28
TOTAL		315	1,148	5,302

REMARKS: (1) Includes metal waste first cycle, section five, and second cycle wastes resulting from cleanouts. ** Reserve calculations based on storage of combined section five and second cycle wastes only at 275 gal. of sludge per run. *** Reserve calculations based on storage of combined section five, second cycle, and 224-T building waste at 350 gal. of sludge per run.

224 BUILDING WASTES

	Cascade In Use	Gallons x 10^3		Estimated Reserve Capacity In Batches
		Cribbed	Sludge	
EAST AREA	201-203-202-B 204-201-T	11 (2) Inactive Cascade	170 187	476
WEST AREA	110-111-112-T	132	Included with 2nd cycle figures	
TOTAL		143	357	476

*REMARKS: Flushes of B plant are cribbed as follows: (1) 2nd cycle, 1st cycle, stack drainage, and section 5 crib via 110-111-112-B tanks. (2) 224-B and Section 9 metal waste tanks crib via 200-B tanks.

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WASTE STORAGE STATUS

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As of 12-31-52

TBP WASTE

PERMANENT AND TEMPORARY WASTE

TANK	Gallons x 10^3							
	In Storage Beg Mo		TBP Received		In Storage End Mo		Reserve Capacity	
BRSF AREA	Perm.	Temp.	Perm.	Temp.	Perm.	Temp.	Perm.	Temp.
[redacted]		(1)		(1)		(1)		(1)
[redacted]	362		198		560		1,362 (4)	
HK							602	
BF		55		11		66	757	
TOTAL	362	55	198		560	66	2,721	
BEST AREA								
[redacted]	426		774				57	
[redacted]							216	
BF		9 (1) 371 (2)				9 (1) 371 (2)		
TX							3,010	
TOTAL	426	380	774	0	1,200	380	3,283	

Supernate tanks, 109 BF and 115 TX not included in the above.

Gallons of TBP of condensate to crib 3,909 (3)

Volumetric Ratios:

	Waste to Feed (Permanent only)	(Combined Perm. and Temp.)
During Month	1.90	1.93
Cumulative to date	1.17	1.47

Feed adjusted to metal waste equivalent

REMARKS: (1) Temporary wastes and reserve capacities are shown with metal waste inventories. (2) Temporary waste stored in 115 TX supernate tank. (3) Cribbed waste contained 1.62 tons of depleted and decontaminated uranium. (4) Tank 109-C is not included in reserve figure since this space is not immediately available.

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WASTE STORAGE STATUS

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As of 12-31-52

WASTE EVAPORATOR

FARM	Gallons x 10^3			
	Feed	Bottoms	Condensate To Crib	Reserve Bottoms Space
EAST AREA				
T		197	475	303
[illegible]				
[illegible]	329			
[illegible]	343			
TOTAL	672	197	475	(1) 303

On the basis of present bottoms reserve figures, it is estimated that a re-evaporation of [illegible] evaporator bottoms tanks will be required near the middle of February, 1953 before further space recovery can be made.

[illegible]				
[illegible]				
TX	—	—	—	676
[illegible]				
TOTAL	—	—	—	676

REMARKS:

REDOX WASTE

FARM	Gallons x 10^3			**
	In Storage Beg. No.	Waste Received	In Storage End No.	Reserve Capacity In Batches
S Farm	3,224	373	3,597	2,103
SU Farm				
TOTAL	3,224	373	3,597	2,103

Gallons x 10^3 of condensate to crib 378.

** Will be considered as Redox reserve capacity when empty.

[redacted]

* Reserve Capacity at 2,600 gallons per 6-3 equivalent batch

REMARKS: A total of 2,713,000 gallons of storage space is available in 241-TY and 241-[illegible] tank farms.

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STATUS OF WASTE TANKS

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	Type of Waste	Gallons x 10^3		Reserve Cap.		Remarks
		Capacity	Stored	Gallons x 10^3	In Batches	
B Farm						
101	MW	530	530			
102	MW	530	530			
103	MW	530	519			
104	RB	530	531			Started filling 12-4-52.
105	RB	530	227	303	RB	
106	IC	530	415	115	42	Evaporator Feed Tank
107	RB	530	531			
108	RB	530	528			
109	RB	530	531			Completed filling 12-4-52
110	IC-2C	5-6	530	530		Active Cascade - B Plant
111	IC-2C	5-6	530	530		
112	IC-2C	5-6	542	542		
201	MW	224	54.5	54.5		Active Cascade - B Plant
202	MW	224	54.5	54.5		
203	MW	224	54.5	54.5		
204	MW	224	54.5	54.5		
C Farm						
101	MW	530				Cascade now processing for feed
102	MW	530	984	595		to TBP Plant
103	MW	530				
104	MW	530	530			
105	MW	530	530			
106	MW	530	519			
107	IC-TBP	530	399-148	(-16)		Overflow line to 108 C plugged
108	IC-TBP	530	34-51	445	TBP	/12-18-52
109	IC-TBP	530	10-486	23	Supernate	Temp. supernate tank for C-Farm
						MW removal operations
110	IC-TBP	530	231-259	40	TBP	Overline still plugged 12-31-52
111	IC-TBP	530	36-103	391	TBP	
112	IC-TBP	530	17-0	502	TBP	
201	MW	54.5	52.5			
202	MW	54.5	54.5			
203	MW	54.5	54.5			
204	MW	54.5	54.5			

MW - Metal Waste
IC - First Cycle Waste
2C - Second Cycle Waste
R - Redox Waste
224 - 224 Bldg. Waste
TBP - TBP Waste
RB - Evaporator Bottoms

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[redacted]-SECURITY INFORMATION

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STATUS OF WASTE TANKS

	Type of Waste	Gallons x 10^3 Capacity	Gallons x 10^3 Stored	Reserve Capacity Gallons x 10^3	Reserve Capacity In Batches	Remarks
BX Farm						
101	MW	530	530			
102	MW	530	467	63	23	
103	MW	530	530			
104	MW	530	530			
105	MW	530	530			
106	MW	530	530			
107	1C-TBP	530	437-0	93	TBP	Finished pumping 12-18-52, down [illegible]
108	1C-TBP	530	21-0	509	TBP	sludge. Finished pumping to 11,00[illegible]
109	1C-TBP	530	530			heel, 1-1-52. 23,000 gallons of [illegible]
110	1C	530	530			1st cycle supernate was lost to [illegible]
111	1C	530	525			open ground from 108 BX pumping [illegible]
112	1C	530	530			operation on 12-16-52.
BY Farm						
101	MW	758	758			
102	MW	758	758			
103	MW	758	664	83	31	
104	MW	758	756			
105	MW	758	491	267	99	
106	MW	758		747	277	
107	1C-TBP	758	1	757	TBP	Pumped to liquid heel - 12-6-52
108	1C	758	753			
109	TBP	758		758	Supernate	TBP supernate tank - 200-E Area
110	1C	758	732	21	[illegible]	
111	1C	758	758			Receives high TBP wastes for tem-
112	MW	758	362	385	143	orary storage.
						(Received 11,000 gallons in Dec.)
S Farm						
101	MW	530	530			
102	MW	580	530			
103	MW	530	519			
104	1C	530	530			1 yr. decay complete 3-31-53
105	1C	530	530			Evaporation to start after 4-1-53
106	1C	530	529			
107	1C-TBP	530	245-285			
108	1C-TBP	530	73-457			Tank filled 12-11-52
109	1C-TBP	530	4-458	57	TBP	Tank now filling with TBP waste.

* MW - Metal Waste
1C - First Cycle Waste
2C - Second Cycle Waste
R - Redox Waste

22h - 22h Bldg. Waste
TBP - TBP Waste
EB - Evaporator Bottoms

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STATUS OF WASTE TANKS

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HW-27

	Type of Waste	Gallons x 10^3 Capacity	Gallons x 10^3 Stored	Reserve Capacity Gallons x 10^3	Reserve Capacity In Batches	Remarks
T Farm						
110 5-6	224A2C	530	530			Active cascade - T Plant
111 5-6	224A2C	530	530			
112 5-6	224A2C	569	569			112-T Cascades to [illegible]
201	224	54.5	54.5			
202	224	54.5	54.5			
203	224	54.5	54.5			
204	224	54.5	54.5			
U Farm						
101	MW	530				Cascade now serving as feed to [illegible] Plant
102	MW	530	797	782		
103	MW	530				
104	MW	530	296	234		Supernate from 104-U pumped to Cascade 101-103U
105	MW	530	530			
106	MW	530	519			
107	MW	530	530			
108	MW	530	530			
109	MW	530	519			
110	1C	530	336	194	75**	Cascade being held as Redox waste reserve.
111	1C	530	14	516	198	
112	1C	530	32	498	192	
201	TBP	54.5		52.5	TBP	
202	TBP	54.5		54.5	TBP	
203	TBP	54.5		54.5	TBP	
204	TBP	54.5		54.5	TBP	
BX Farm						
101	MW	758	758			
102	MW	758	758			
103	MW	758	758			
104	MW	758	750			
105	MW	758	766			
106	MW	758	292	466	173	Active cascade 221-T Metal Waste
107	MW	758	766			
108	MW	758	751			
109	1C	758	758			Active cascade 221-T - 1st Cycle
110	1C	758	398	360	129	
111	1C	758		758	271	
112	1C	758		753	269	Tank to be held for Recuplex waste

* MW - Metal Waste
* 1C - First Cycle Waste
2C - Second Cycle
R - Redox Waste
224 - 224 Bldg. Waste
TBP - TBP Waste
EB - Evaporator Bottoms
* See bottom of page H-2902.11-I

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STATUS OF WASTE FARMS

	Type of Waste	Gallons x 10^3 Capacity	Gallons x 10^3 Stored	Reserve Capacity Gallons x 10^3	Reserve Capacity In Batches	Remarks
TX Farm (cont'd)						
113	EB	758	748	10	EB	
114	MW	758	666	81	30	
115	TBP	758	759			TBP supernate tank, 200-W Area
116	EB	758	92	666	EB	Active bottoms tank, 200-W Area
117	EB	758	756			
118	1C	758	718	40	14	Evap. feed tank, 200-W Area
TY Farm						
101	R	758		758	291 **	
102	R	758		747	287	
103	TBP	758		758	TBP	
104	TBP	758		747	TBP	
105	TBP	758		758	TBP	
106	TBP	758		747	TBP	
S Farm						
101	R	758		758	291 **	
102	R	758		758	291	
103	R	758		747	287	
104	R	758		758	291	
105	R	758		758	291	
106	R	758		747	287	
107	R	758	763			
108	R	758	760			Filled 12-24-52; cascaded to 109-S. Active tank, Redox Wastes
109	R	758	103	644	248	
110	R	758	774			
111	R	758	755			
112	R	758	442	305	117	

* MW - Metal Waste
1C - First Cycle Waste
2C - Second Cycle Waste
R - Redox Waste

224 & 224 Bldg. Waste
TBP - TBP Waste
EB - Evaporator Bottoms
** Based on Plant performance of 2,660 gallons per batch - (average - Oct, Nov., 1952)

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APPENDIX B

WHC-MR-0132, A HISTORY OF THE 200 AREA TANK FARMS

TANKS 241-B-201 THROUGH 241-B-204

AND

TANKS 241-T-201 THROUGH 241-T-204

201-B-1 WHC-MR-0132

Waste Status Summary of 201-B Tank-Capacity 55,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid In Storage	Solids In Storage	Remarks
1-1944					
2					
3					
4					
1-1945					
2					
3					
4					
1-1946					
2					
3					
4					
1-1947					
2					
3					
4					
1-1948					
2					
3					
4					
1-1949					
2					
3					
4					
1-1950					
2					
3					
4					
1-1951					
2					
3					
4					
1-1952	224	54.5	---	---	Active cascade to crib
2	224	54.5	---	---	Active cascade to crib
3	224	54.5	---	---	Active cascade to crib
4	224-MW	54.5	---	---	Active cascade to crib

201-B-2 WHC-MR-0132

Waste Status Summary of 201-B Tank-Capacity 55,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid In Storage	Solids In Storage	Remarks
1-1953	224-MW	54.5	---	---	Receives B plant flushes
2	224-MW	54.5	0	54.5	
3	224-MW	54.5	0	54.5	
4	224	54.5	0	54.5	
1-1954	224	54.5	0	54.5	
2	224	54.5	0	54.5	
3	224	54.5	0	54.5	
4	224	54.5	0	54.5	
1-1955	224	54.5	0	54.5	
2	224	54.5	0	54.5	
3	224	54.5	0	54.5	
4	224	54.5	0	54.5	
1-1956	224	54.5	0	54.5	
2	224	54.5	0	54.5	
3	224	54.5	0	54.5	
4	224	54.5	0	54.5	
1-1957	224	53	0	54.5	Latest electrode reading
2	224	53	0	54.5	
3	224	53	24.5	28.5	
4	224	53	24.5	28.5	
1-1958	224	53	24.5	28.5	
2	224	53	24.5	28.5	
3	224	53	29.5	23.5	
4	224	51	22.5	28.5	Latest electrode reading
1-1959	224	51	22.5	28.5	
2	224	51	22.5	28.5	New electrode
3	224	51	22.5	28.5	
4	224	52	0	54.5	
1-1960	224	52	0	54.5	
2	224	52	0	54.5	
3	224	52	0	54.5	
4	224	52	0	54.5	
1-1961	224	---	0	54.5	
2	224	50	0	54.5	
3	224	50	0	54.5	
4	224	50	0	54.5	

201-B-3 WHC-MR-0132

Waste Status Summary of 201-B Tank-Capacity 55,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid In Storage	Solids In Storage	Remarks
1-1962	224	50	0	54.5	
2	224	50	0	54.5	
3	224	---	---	---	
4	224	51	1	50	
1-1963	224	51	1	50	
2	224	51	1	50	
3	224	---	---	50	
4	224	53	3	50	Latest electrode reading
1-1964	224	53	3	50	
2	224	53	3	50	
3	224	53	3	50	
4	224	53	3	50	
1-1965	---	---	---	50	
2	224	56	6	50	
3	224	56	6	50	
4	224	56	6	50	
1-1966	224	56	6	50	
2	224	56	6	50	
3	224	56	6	50	
4	224	56	6	50	
1-1967	224	56	6	50	
2	224	56	6	50	
3	224	56	6	50	
4	224	56	6	50	
1-1968	224	55	5	50	
2	224	55	5	50	
3	224	55	5	50	
4	224	55	5	50	
1-1969	224	55	5	50	
2	224	55	5	50	
3	224	55	5	50	
4	224	55	5	50	
1-1970	224	55	25	30	
2	224	55	25	30	
3	224	55	25	30	
4	224	54	24	30	

201-B-4 WHC-MR-0132

Waste Status Summary of 201-B Tank-Capacity 55,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid In Storage	Solids In Storage	Remarks
1-1971	224	54	24	30	
2	224	33	3	30	22 to 106-B
3	224	33	3	30	
4	224	33	3	30	
1-1972	224	33	7	26	
2	224	33	7	26	
3	224	33	7	26	
4	224	33	7	26	
1-1973	224	33	7	26	Suspect leaker
2	224	33	7	26	Suspect leaker
3	224	33	7	26	Suspect leaker
4	224	33	7	26	Suspect leaker
1-1974	224	32	6	26	Suspect leaker, 1 to 109-B
2	224	31	5	26	Suspect leaker, 2 to 109-B
3	224	29	3	26	Suspect leaker, 4 to 109-B
4	---	29	0	29	Suspect leaker, 4 water, 6 to 109-B
1-1975	---	29	0	29	Suspect leaker, 2 to 102-B
2	---	29	0	29	Removed from service, 1 to 109-B
3	---	29	0	29	" " "
4	---	29	0	29	" " "
1-1976	---	29	0	29	" " "
2	---	29	0	29	
3	---	29	0	29	
4	---	29	0	29	Salt Well Comp.
1-1977	---	29	0	29	Questionable Integrity
2	---	29	0	29	
3	---	29	0	29	Inactive-Stabilized
4	---	29	0	29	Inactive Current-Stabilized Phase I

201-B-5 WHC-MR-0132

Waste Status Summary of 201-B Tank-Capacity 530,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid in Storage	Solids in Storage	Remarks
1-1978	-	29	0	29	Inactive - Primary Stabilized
2-	-	29	0	29	
3-	-	29	0	29	P-10 Pmp. Removed
4-	-	29	0	29	
1-1979	-	28	1	27	New Solids Level 1/29/79
2-	-	28	1	27	Questionable Integrity
3-	-	28	1	27	
4-	-	28	1	27	
1-1980	-	28	1	27	New Photo 2/4/80
2-	-	28	1	27	
3-	-	29	1	28	
4-	-	29	1	28	

202-B-I WHC-MR-0132

Waste Status Summary of 202-B Tank-Capacity 55,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid In Storage	Solids In Storage	Remarks
1-1944					
2					
3					
4					
1-1945					
2					
3					
4					
1-1946					
2					
3					
4					
1-1947					
2					
3					
4					
1-1948					
2					
3					
4					
1-1949					
2					
3					
4					
1-1950					
2					
3					
4					
1-1951					
2					
3					
4					
1-1952	224	54.5	---	---	Active cascade to crib
2	224	54.5	---	---	Active cascade to crib
3	224	54-5	---	---	Active cascade to crib
4	224-MW	54.5	---	---	Active cascade to crib

202-B-2 WHC-MR-0132

Waste Status Summary of 202-B Tank-Capacity 55,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid In Storage	Solids In Storage	Remarks
1-1953	224-MW	54.5	---	---	Receives B plant flushes
2	224-MW	54.5	---	54.5	
3	224-MW	54.5	29.5	25.0	
4	224	54.5	29.5	25.0	
1-1954	224	54.5	29.5	25.0	
2	224	54.5	29.5	25.0	
3	224	54.5	29.5	25.0	Cascades to crib
4	224	54.5	29.5	25.0	Rec'd 5-6 water. Cascades to crib
1-1955	224	54.5	29.5	25.0	
2	224	54.5	29.5	25.0	Cascades to crib
3	224	54.5	29.5	25.0	Rec'd 224-B flush water. Cascades tc crib
4	224	54.5	29.5	25.0	
1-1956	224	54.5	29.5	25.0	
2	224	54.5	29.5	25.0	
3	224	54.5	29.5	25.0	
4	224	54.5	29.5	25.0	
1-1957	224	56	31	25.0	Latest electrode reading
2	224	56	31	25.0	
3	224	56	31	25.0	
4	224	56	31	25.0	
1-1958	224	56	31	25	
2	224	56	31	25	
3	224	56	31	25	
4	224	54	29	25	Latest electrode reading
1-1959	224	54	29	25	
2	224	54	29	25	New electrode
3	224	54	29	25	
4	224	54	29	25	
1-1960	224	54	29	25	
2	224	54	29	25	
3	224	51	26	25	
4	224	51	26	25	
1-1961	224	51	26	25	
2	224	51	26	25	
3	224	51	26	25	
4	224	51	26	25	

202-B-3 WHC-MR-0132

Waste Status Summary of 202-B Tank-Capacity 55,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid In Storage	Solids In Storage	Remarks
1-1962	224	---	---	25	
2	224	54.5	29.5	25	7.5 from 221-B
3	224	---	---	25	
4	224	55	30	25	
1-1963	224	55	30	25	
2	224	55	30	25	
3	224	---	---	25	
4	224	54	29	25	Latest electrode reading
1-1964	224	54	29	25	
2	224	54	29	25	
3	224	54	29	25	
4	224	54	29	25	
1-1965	---	---	---	25	
2	224	58	33	25	
3	224	56	31	25	
4	224	56	31	25	
1-1966	224	56	31	25	
2	224	56	31	25	
3	224	56	31	25	
4	224	56	31	25	
1-1967	224	56	31	25	
2	224	56	31	25	
3	224	56	31	25	
4	224	56	31	25	
1-1968	224	56	31	25	
2	224	56	31	25	
3	224	56	31	25	
4	224	56	31	25	
1-1969	224	56	31	25	
2	224	56	31	25	
3	224	56	31	25	
4	224	56	31	25	
1-1970	224	56	27	29	
2	224	56	27	29	
3	224	56	27	29	
4	224	56	27	29	

202-B-4 WHC-MR-0132

Waste Status Summary of 202-B Tank-Capacity 55,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid In Storage	Solids In Storage	Remarks
1-1971	224	56	27	29	
2	224	56	27	29	
3	224	56	27	29	
4	224	56	27	29	
1-1972	224	56	29	27	
2	224	56	29	27	
3	224	56	29	27	
4	224	56	29	27	
1-1973	224	56	29	27	
2	224	56	29	27	
3	224	56	29	27	
4	224	56	29	27	
1-1974	224	53	26	27	3 to 109-B
2	224	53	26	27	
3	224	53	26	27	
4	224	53	26	27	
1-1975	224	53	26	27	
2	224	53	26	27	
3	224	53	26	27	
4	224	53	26	27	
1-1976	224	53	26	27	
2	224	53	26	27	
3	---	53	26	27	Restricted
4	---	53	26	27	"
1-1977	---	53	26	27	Restricted
2	---	30	3	27	"
3	---	27	0	27	Inactive Current
4	---	27	0	27	" "

202-B-5 WHC-MR-0132

Waste Status Summary of 202-B Tank-Capacity 530,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid in Storage	Solids in Storage	Remarks
1-1978	-	27	0	27	Inactive
2-	-	27	0	27	
3-	-	27	0	27	
4-	-	27	0	27	
1-1979	-	27	0	27	
2-	-	27	0	27	
3-	-	27	0	27	
4-	-	27	0	27	
1-1980	-	27	0	27	New Photo 2/4/80
2-	-	27	0	27	
3-	-	28	0	28	
4-	-	28	0	28	

203-B-1 WHC-MR-0132

Waste Status Summary of 203-B Tank-Capacity 55,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid In Storage	Solids In Storage	Remarks
1-1944					
2					
3					
4					
1-1945					
2					
3					
4					
1-1946					
2					
3					
4					
1-1947					
2					
3					
4					
1-1948					
2					
3					
4					
1-1949					
2					
3					
4					
1-1950					
2					
3					
4					
1-1951					
2					
3					
4					
1-1952	224	54.5	---	---	Active cascade to crib
2	224	54.5	---	---	Active cascade to crib
3	224	54.5	---	---	Active cascade to crib
4	224-MW	54.5	---	---	Active cascade to crib

203-B-2 WHC-MR-0132

Waste Status Summary of 203-B Tank-Capacity 55,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid In Storage	Solids In Storage	Remarks
1-1953	224-MW	54.5	---	---	Receives B plant flushes
2	224-MW	54.5	0	54.5	
3	224-MW	54.5	0	54.5	
4	224	54.5	0	54.5	
1-1954	224	54.5	0	54.5	
2	224	54.5	0	54.5	
3	224	54.5	0	54.5	
4	224	54.5	0	54.5	
1-1955	224	54.5	0	54.5	
2	224	54.5	0	54.5	
3	224	54.5	0	54.5	
4	224	54.5	0	54.5	
1-1956	224	54.5	0	54.5	
2	224	54.5	0	54.5	
3	224	54.5	0	54.5	
4	224	54.5	0	54.5	
1-1957	224	56	1.5	54.5	Latest electrode reading
2	224	56	1.5	54.5	
3	224	56	1.5	54.5	
4	224	56	1.5	54.5	
1-1958	224	56	1.5	54.5	
2	224	56	1.5	54.5	
3	224	56	1.5	54.5	
4	224	55	0.5	54.5	Latest electrode reading
1-1959	224	55	0.5	54.5	
2	224	55	0.5	54.5	
3	224	55	0.5	54.5	
4	224	55	0.5	54.5	
1-1960	224	55	0.5	54.5	
2	224	55	0.5	54.5	
3	224	55	0.5	54.5	
4	224	55	0.5	54.5	
1-1961	224	---	---	54.5	
2	224	54	0	54.5	
3	224	54	0	54.5	
4	224	54	0	54.5	

203-B-3 WHC-MR-0132

Waste Status Summary of 203-B Tank-Capacity 55,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid In Storage	Solids In Storage	Remarks
1-1962	224	---	---	54.5	
2	224	56	1.5	54.5	
3	224	56	---	---	
4	224	56	2	54	
1-1963	224	56	2	54	
2	224	56	2	54	
3	224	56	2	54	
4	224	56	2	54	Latest electrode reading
1-1964	224	---	---	54	
2	224	55	1	54	New electrode
3	224	55	1	54	
4	224	55	1	54	
1-1965	---	---	---	54	
2	224	58	4	54	
3	224	56	2	54	
4	224	56	2	54	
1-1966	224	56	2	54	
2	224	56	2	54	
3	224	56	2	54	
4	224	56	2	54	
1-1967	224	56	2	54	
2	224	56	2	54	
3	224	56	2	54	
4	224	56	2	54	
1-1968	224	56	2	54	
2	224	56	2	54	
3	224	56	2	54	
4	224	56	2	54	
1-1969	224	56	2	54	
2	224	56	2	54	
3	224	56	2	54	
4	224	56	2	54	
1-1970	224	56	7	49	
2	224	56	7	49	
3	224	56	7	49	
4	224	56	7	49	

203-B-4 WHC-MR-0132

Waste Status Summary of 203-B Tank-Capacity 55,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid In Storage	Solids In Storage	Remarks
1-1971	224	56	7	49	
2	224	56	7	49	
3	224	56	7	49	
4	224	56	7	49	
1-1972	224	56	12	44	
2	224	56	12	44	
3	224	56	12	44	
4	224	56	12	44	
1-1973	224	56	12	44	
2	224	56	12	44	
3	224	56	12	44	
4	224	56	12	44	
1-1974	224	50	6	44	6 to 109-B
2	224	50	6	44	
3	224	50	6	44	
4	224	50	6	44	
1-1975	224	50	6	44	
2	224	50	6	44	
3	224	50	6	44	
4	224	50	6	44	
1-1976	224	50	6	44	
2	224	50	6	44	
3	---	50	5	45	Restricted
4	---	50	5	45	"
1-1977	---	50	5	45	Restricted
2	---	50	5	45	"
3	---	50	3	47	Inactive Current-Solid Level Adj.
4	---	50	3	47	" " " " "

203-B-5 WHC-MR-0132

Waste Status Summary of 203-B Tank-Capacity 530,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid in Storage	Solids in Storage	Remarks
1-1978	-	50	3	47	Inactive
2-	NCPLX	50	3	47	
3-	NCPLX	50	3	47	
4-	NCPLX	50	3	47	
1-1979	NCPLX	50	3	47	New Photo's 3/1/79
2-	NCPLX	50	3	47	
3-	NCPLX	50	3	47	
4-	NCPLX	50	3	47	
1-1980	NCPLX	50	3	47	New Photo 2/6/80
2-	NCPLX	50	3	47	
3-	NCPLX	50	2	48	
4-	NCPLX	50	2	48	

204-B-1 WHC-MR-0132

Waste Status Summary of 204-B Tank-Capacity 55,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid In Storage	Solids In Storage	Remarks
1-1944					
2					
3					
4					
1-1945					
2					
3					
4					
1-1946					
2					
3					
4					
1-1947					
2					
3					
4					
1-1948					
2					
3					
4					
1-1949					
2					
3					
4					
1-1950					
2					
3					
4					
1-1951					
2					
3					
4					
1-1952	224	54.5	---	---	Active cascade to crib
2	224	54.5	---	---	Active cascade to crib
3	224	54.5	---	---	Active cascade to crib
4	224-MW	54.5	---	---	Active cascade to crib

204-B-2 WHC-MR-0132

Waste Status Summary of 204-B Tank-Capacity 55,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid In Storage	Solids In Storage	Remarks
1-1953	224-MW	54.5	---	---	Active cascade to crib
2	224-MW	54.5	0	54.5	
3	224-MW	54.5	0	54.5	
4	224	54.5	0	54.5	
1-1954	224	54.5	0	54.5	
2	224	54.5	0	54.5	
3	224	54.5	0	54.5	
4	224	54.5	0	54.5	
1-1955	224	54.5	0	54.5	
2	224	54.5	0	54.5	
3	224	54.5	0	54.5	
4	224	54.5	0	54.5	
1-1956	224	54.5	0	54.5	
2	224	54.5	0	54.5	
3	224	54.5	0	54.5	
4	224	54.5	0	54.5	
1-1957	224	56	1.5	54.5	Latest electrode reading
2	224	56	1.5	54.5	
3	224	56	1.5	54.5	
4	224	56	1.5	54.5	
1-1958	224	56	1.5	54.5	
2	224	56	1.5	54.5	
3	224	56	1.5	54.5	
4	224	55	0.5	54.5	Latest electrode readings
1-1959	224	56	1.5	54.5	Latest electrode reading
2	224	56	1.5	54.5	New electrode
3	224	56	1.5	54.5	
4	224	54	0	54	Latest electrode reading
1-1960	224	54	0	54	New electrode installed
2	224	54	0	54	
3	224	54	0	54	
4	224	54	0	54	
1-1961	224	54	0	54	
2	224	54	0	54	
3	224	54	0	54	
4	224	54	0	54	

204-B-3 WHC-MR-0132

Waste Status Summary of 204-B Tank-Capacity 55,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid In Storage	Solids In Storage	Remarks
1-1962	224	---	---	54.5	
2	224	56	1.5	54.5	
3	224	56	---	54	
4	224	56	2	54	
1-1963	224	56	2	54	
2	224	56	2	54	
3	224	56	2	54	
4	224	56	2	54	Latest electrode reading
1-1964	224	---	---	54	
2	224	55	1	54	New electrode
3	224	55		54	
4	224	55		54	
1-1965	---	---	---	54	
2	224	58	4	54	
3	224	56	2	54	
4	224	56	2	54	
1-1966	224	56	2	54	
2	224	56	2	54	
3	224	56	2	54	
4	224	56	2	54	
1-1967	224	56	2	54	
2	224	56	2	54	
3	224	56	2	54	
4	224	56	2	54	
1-1968	224	56	2	54	
2	224	56	2	54	
3	224	56	2	54	
4	224	56	2	54	
1-1969	224	56	2	54	
2	224	56	2	54	
3	224	56	2	54	
4		56	2	54	
1-1970	224	56	8	48	
2	224	56	8	48	
3	224	56	8	48	
4	224	56	8	48	

204-B-4 WHC-MR-0132

Waste Status Summary of 204-B Tank-Capacity 55,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid In Storage	Solids In Storage	Remarks
1-1971	224	56	8	48	
2	224	56	8	48	
3	224	56	8	48	
4	224	56	8	48	
1-1972	224	56	10	46	
2	224	56	10	46	
3	224	56	10	46	
4	224	56	10	46	
1-1973	224	56	10	46	
2	224	56	10	46	
3	224	56	10	46	
4	224	56	10	46	
1-1974	224	49	3	46	6 to 109-B
2	224	49	3	46	
3	224	49	3	46	
4	224	49	3	46	
1-1975	224	49	3	46	
2	224	49	3	46	
3	224	49	3	46	
4	224	49	3	46	
1-1976	224	49	3	46	Removed from Service
2	224	49	3	46	
3	---	49	3	46	Restricted
4	---	49	3	46	"
1-1977	---	49	3	46	Restricted
2	---	49	3	46	"
3	---	49	3	46	Inactive Current
4	---	49	3	46	" "

204-B-5 WHC-MR-0132

Waste Status Summary of 204-B Tank-Capacity 530,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid in Storage	Solids in Storage	Remarks
1-1978	-	49	3	46	Inactive
2-	NCPLX	49	3	46	
3-	NCPLX	49	3	46	
4-	NCPLX	49	3	46	
1-1979	NCPLX	49	3	46	
2-	NCPLX	49	3	46	
3-	NCPLX	49	3	46	
4-	NCPLX	49	3	46	New Photo 10/16/79
1-1980	NCPLX	49	3	46	
2-	NCPLX	49	3	46	
3-	NCPLX	50	3	47	
4-	NCPLX	50	3	47	

201-T-1 WHC-MR-0132

Waste Status Summary of 201-T Tank-Capacity 55,000 Gallons.

Qtr.-Year	Type Waste	Total Vol.	Liquid In Storage	Solids In Storage	Remarks
1-1952	---	---	---	---	
2	224	54.5	---	---	These 4 tanks out of service 5/29/52.
3	224	54.5	---	---	
4	224	54.5	---	---	
1-1953	---	54.5	---	54.5	
2	---	54.5	0	54.5	
3	---	54.5	0	54.5	
4	---	54.5	0	54.5	
1-1954	---	54.5	0	54.5	
2	---	54.5	0	54.5	
3	---	54.5	0	54.5	
4	---	54.5	0	54.5	
1-1955	---	54.5	0	54.5	
2	---	54.5	0	54.5	
3	---	54.5	0	54.5	
4	---	54.5	0	54.5	
1-1956	---	54.5	0	54.5	
2	---	54.5	0	54.5	
3	---	54.5	0	54.5	
4	---	54.5	0	54.5	
1-1957	---	54.5	0	54.5	Estimated reading.
2	---	54.5	0	54.5	New electrode reading.
3	---	54		54.5	
4	---	54		54.5	
1-1958	224	54	.5	54.5	
2	224	54	.5	54.5	
3	224	54	.5	54.5	
4	224	54	.5	54.5	
1-1959	224	54	.5	54.5	
2	224	54	.5	54.5	
3	224	54	.5	54.5	
4	224	54	.5	54.5	

201-T-2 WHC-MR-0132

Waste Status Summary of 201-T Tank-Capacity 55,0000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid In Storage	Solids In Storage	Remarks
1-1960	224	54	.5	54.5	
2	224	54	5	54.5	
3	224	54	.5	54.5	
4	224	54	.5	54.5	
1-1961	---	---	---	---	6 Month Report
2	224	54	.5	54.5	
3	---	---	---	---	
4	224	54	.5	54.5	6 Month Report
1-1962	---	---	---	---	
2	224	53	---	53	6 Month Report
3	---	---	---	---	
4	224	52	2	50	Latest electrode readings.
1-1963	---	---	---	---	6 Month Report
2	224	54	4	50	New electrode reading.
3	---	---	---	---	
4	224	54	4	50	6 Month Report
1-1964	---	---	---	---	
2	224	54	4	50	6 Month Report
3	---	---	---	---	
4	224	54	4	50	6 Month Report
1-1965	---	---	---	---	6 Month Report
2	224	53	20	33	
3	224	53	20	33	
4	224	53	20	33	
1-1966	224	53	20	33	
2	224	53	20	33	
3	224	53	20	33	
4	224	53	20	33	
1-1967	224	53	20	33	
2	224	53	20	33	
3	224	53	20	33	
4	224	53	20	33	
1-1968	224	48	15	33	
2	224	48	15	33	
3	224	48	15	33	
4	224	49	16	33	

201-T-3 WHC-MR-0132

Waste Status Summary of 201-T Tank-Capacity 55,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid In Storage	Solids In Storage	Remarks
1-1969	224	53	20	33	
2	224	54	21	33	
3	224	54	21	33	
4	224	54	21	33	
1-1970	224	54	21	33	
2	224	54	21	33	
3	224	54	21	33	
4	224	54	23	31	
1-1971	224	54	23	31	
2	224	54	23	31	
3	224	54	23	31	
4	224	54	23	31	
1-1972	224	54	23	31	
2	224	54	23	31	
3	224	54	23	31	
4	224	54	23	31	
1-1973	224	54	23	31	
2	224	54	23	31	
3	224	54	23	31	
4	224	54	23	31	
1-1974	224	54	23	31	
2	224	54	23	31	
3	224	54	23	31	
4	224	54	23	31	
1-1975	224	54	23	31	
2	224	54	23	31	
3	224	54	23	31	
4	224	54	23	31	
1-1976	224	34	3	31	Removed from service 21 to 101-T
2	224	33	2	31	" " " 1 to 101-T.
3	Evap.	31	0	31	Inactive salt well pumping.
4	---	31	0	31	
1-1977	---	31	0	31	Salt Well, Pump
2	---	31	0	31	Salt Well, Pump
3	---	31	0	31	Inactive Current
4	---	31	0	31	" " " Salt Well installed

201-T-4 WHC-MR-0132

Waste Status Summary of 201-T Tank-Capacity 55,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid in Storage	Solids in Storage	Remarks
1-1978	-	28	0	28	Photo taken 2-27-78
2-	-	27	0	27	Prim. Stabilized New Solids Level Adj. 5-31-78
4-	-	27	0	27	
1-1979	-	27	0	27	
2-	-	27	0	27	
3-	-	27	0	27	
4-	-	27	0	27	
1-1980	-	27	0	27	New Photo 3-10-80
2-	-	27	0	27	
3-	-	28	0	28	
4-	-	28	0	28	

202-T-1 WHC-MR-0132

Waste Status Summary of 202-T Tank-Capacity 55,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid In Storage	Solids In Storage	Remarks
1-1952	---	---	---	---	
2	224	54.5	---	---	
3	224	54.5	---	---	
4	224	54.5	---	---	
1-1953	---	54.5	---	54.5	
2	---	54.5	0	54.5	
3	---	54.5	0	54.5	
4	---	54.5	0	54.5	
1-1954	---	54.5	0	54.5	
2	---	54.5	0	54.5	
3	---	54.5	0	54.5	
4	---	54.5	0	54.5	
1-1955	---	54.5	0	54.5	
2	---	54.5	0	54.5	
3	---	54.5	0	54.5	
4	---	54.5	0	54.5	
1-1956	---	54.5	0	54.5	
2	---	54.5	0	54.5	
3	---	54.5	0	54.5	
4	---	54.5	0	54.5	
1-1957	---	54.5	0	54.5	Estimated reading.
2	---	54.5	0	54.5	New electrode reading.
3	224	55	.5	54.5	Latest electrode reading.
4	224	55	.5	54.5	
1-1958	224	55	.5	54.5	
2	224	55	.5	54.5	
3	224	55	.5	54.5	
4	224	55	.5	54.5	
1-1959	224	56	1.5	54.5	
2	224	55	.5	54.5	
3	224	55	.5	54.5	
4	224	55	.5	54.5	

202-T-2 WHC-MR-0132

Waste Status Summary of 202-T Tank-Capacity 55,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid In Storage	Solids In Storage	Remarks
1-1960	224	55	.5	54.5	
2	224	55	.5	54.5	
3	224	55	.5	54.5	
4	224	55	.5	54.5	
1-1961	---	---	---	---	
2	224	55	.5	54.5	6 Month Report
3	---	---	---	---	" " "
4	224	55	.5	54.5	
1-1962	---	---	---	---	" " "
2	224	54	---	54	
3	---	---	---	---	
4	224	53	3	50	Latest electrode readings.
1-1963	---	---	---	---	
2	224	53	3	50	New electrode reading.
3	---	---	---	---	6 Month Report
4	224	53	3	50	
1-1964	---	---	---	---	" " " "
2	224	53	3	50	
3	---	---	---	---	" " " "
4	224	53	3	50	
					" " " "
1-1965	---	---	---	---	
2	224	52	22	30	
3	224	52	22	30	
4	224	52	22	30	
1-1966	224	52	22	30	
2	224	52	22	30	
3	224	52	22	30	
4	224	52	22	30	
1-1967	224	52	22	30	
2	224	52	22	30	
3	224	52	22	30	
4	224	52	22	30	
1-1968	224	51	21	30	
2	224	51	21	30	
3	224	51	21	30	
4	224	51	21	30	

202-T-3 WHC-MR-0132

Waste Status Summary of 202-T Tank-Capacity 55,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid In Storage	Solids In Storage	Remarks
1-1969	224	51	21	30	
2	224	51	21	30	
3	224	51	21	30	
4	224	51	21	30	
1-1970	224	51	21	30	
2	224	51	21	30	
3	224	51	21	30	
4	224	51	21	30	
1-1971	224	51	21	30	
2	224	51	21	30	
3	224	51	21	30	
4	224	51	21	30	
1-1972	224	51	21	30	
2	224	51	21	30	
3	224	51	21	30	
4	224	51	21	30	
1-1973	224	51	21	30	
2	224	51	21	30	
3	224	51	21	30	
4	224	51	21	30	
1-1974	224	51	21	30	
2	224	51	21	30	
3	224	51	21	30	
4	224	51	21	30	
1-1975	224	51	21	30	
2	224	51	21	30	
3	224	51	21	30	
4	224	51	21	30	
1-1976	---	25	0	25	Removed from service 27 to 101-T
2	---	25	0	25	" " " 1 to 101-T.
3	---	25	0	25	Inactive salt well pumping.
4	---	25	0	25	
1-1977	---	25	0	25	Salt Well, Pump
2	---	25	0	25	Inactive, current
3	---	25	0	25	" " "
4	---	25	0	25	" " " Salt Well Installed

202-T-4 WHC-MR-0132

Waste Status Summary of 202-T Tank-Capacity 50,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid in Storage	Solids in Storage	Remarks
1-1978	-	20	0	20	Prim. Stabl., Photo taken 2-27-78
2-	-	20	0	20	Solids Level taken 1-31-
3-	-	20	0	20	
4-	-	20	0	20	
1-1979	-	20	0	20	
2-	-	20	0	20	
3-	-	20	0	20	
4-	-	20	0	20	New Photo 10-12-79
1-1980	-	20	0	20	
2-	-	20	0	20	
3-	-	21	0	21	
4-	-	21	0	21	

203-T-1 WHC-MR-0132

Waste Status Summary of 203-T Tank-Capacity 55,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid In Storage	Solids In Storage	Remarks
1-1952	---	---	---	---	
2	224	54.5	---	---	
3	224	54.5	---	---	
4	224	54.5	---	---	
1-1953	---	54.5	---	54.5	
2	---	54.5	0	54.5	
3	---	54.5	0	54.5	
4	---	54.5	0	54.5	
1-1954	---	54.5	0	54.5	
2	---	54.5	0	54.5	
3	---	54.5	0	54.5	
4	---	54.5	0	54.5	
1-1955	---	54.5	0	54.5	
2	---	54.5	0	54.5	
3	---	54.5	0	54.5	
4	---	54.5	0	54.5	
1-1956	---	54.5	0	54.5	
2	---	54.5	0	54.5	
3	---	54.5	0	54.5	
4	---	54.5	0	54.5	
1-1957	---	54.5	0	54.5	Estimated reading.
2	---	54.5	0	54.5	New electrode reading.
3	224	55	.5	54.5	Latest electrode reading.
4	224	55	.5	54.5	
1-1958	224	55	.5	54.5	
2	224	55	.5	54.5	
3	224	55	0	55	
4	224	55	.5	54.5	
1-1959	224	55	.5	54.5	
2	224	55	.5	54.5	
3	224	55	.5	54.5	
4	224	55	.5	54.5	

203-T-2 WHC-MR-0132

Waste Status Summary of 203-T Tank-Capacity 55,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid In Storage	Solids In Storage	Remarks
1-1960	224	55	.5	54.5	
2	224	55	.5	54.5	
3	224	55	.5	54.5	
4	224	55	.5	54.5	
1-1961	---	---	---	---	
2	224	55	.5	54.5	6 Month Report
3	---	---	---	---	" " " "
4	224	55	.5	54.5	
1-1962	---	---	---	---	
2	224	54	---	54	" " " "
3	---	---	---	---	" " " "
4	224	53	3	50	Latest electrode reading.
1-1963	---	---	---	---	
2	224	53	3	50	New electrode reading.
3	---	---	---	---	
4	224	53	3	50	6 Month Report
1-1964	---	---	---	---	
2	224	53	3	50	" " " "
3	---	---	---	---	
4	224	53	3	50	" " " "
1-1965	---	---	---	---	" " " "
2	224	52	17	35	
3	224	52	17	35	
4	224	52	17	35	
1-1966	224	52	17	35	
2	224	52	17	35	
3	224	52	17	35	
4	224	52	17	35	
1-1967	224	52	17	35	
2	224	52	17	35	
3	224	52	17	35	
4	224	51	16	35	
1-1968	224	47	12	35	
2	224	51	16	35	
3	224	51	16	35	
4	224	51	16	35	

203-T-3 WHC-MR-0132

Waste Status Summary of 203-T Tank-Capacity 55,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid In Storage	Solids In Storage	Remarks
1-1969	224	51	16	35	
2	224	51	16	35	
3	224	51	16	35	
4	224	51	16	35	
1-1970	224	51	16	35	
2	224	51	16	35	
3	224	51	16	35	
4	224	51	16	35	
1-1971	224	51	16	35	
2	224	51	16	35	
3	224	51	16	35	
4	224	51	16	35	
1-1972	224	51	16	35	
2	224	51	16	35	
3	224	51	16	35	
4	224	51	16	35	
1-1973	224	51	16	35	
2	224	51	16	35	
3	224	51	16	35	
4	224	51	16	35	
1-1974	224	51	16	35	
2	224	51	16	35	
3	224	51	16	35	
4	224	51	16	35	
1-1975	224	51	17	35	
2	224	51	17	35	
3	224	51	17	35	
4	224	51	17	35	
1-1976	224	50	15	35	Removed from service 1 to 101-T.
2	224	42	7	35	" " " 1 to 101-T.
3	Evap.	41	6	35	Inactive salt well pumping.
4	Evap.	40	5	35	
1-1977	Evap.	39	4	35	Salt Well, Pumping
2	Evap.	38	3	35	" " " "
3	Evap.	36	3	35	" " " "
4	Evap.	36	1	35	Salt Well Installed

203-T-4 WHC-MR-0132

Waste Status Summary of 203-T Tank-Capacity 55,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid in Storage	Solids in Storage	Remarks
1-1978	-	37	0	37	Salt Well Installed New Solids Level 1-31-78
2-	-	36	0	36	
3-	-	36	0	36	
4-	-	36	0	36	
1-1979	-	36	0	36	Inactive Primary Stab.
2-	-	36	0	36	
3-	-	36	0	36	
4-	-	36	0	36	
1-1980	-	36	0	36	New Photo 3-18-80
2-	-	36	0	36	
3-	-	37	0	37	
4-	-	37	0	37	

204-T-1 WHC-MR-0132

Waste Status Summary of 204-T Tank-Capacity 55,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid In Storage	Solids In Storage	Remarks
1-1952	---	---	---	---	
2	224	54.5	---	---	
3	224	54.5	---	---	
4	224	54.5	---	---	
1-1953	---	54.5	0	54.5	
2	---	54.5	0	54.5	
3	---	54.5	0	54.5	
4	---	54.5	0	54.5	
1-1954	---	54.5	0	54.5	
2	---	54.5	0	54.5	
3	---	54.5	0	54.5	
4	---	54.5	0	54.5	
1-1955	---	54.5	0	54.5	
2	---	54.5	0	54.5	
3	---	54.5	0	54.5	
4	---	54.5	0	54.5	
1-1956	---	54.5	0	54.5	
2	---	54.5	0	54.5	
3	---	54.5	0	54.5	
4	---	54.5	0	54.5	
1-1957	---	54.5	0	54.5	Estimated reading.
2	---	54.5	0	54.5	New electrode reading.
3	224	56	1.5	54.5	Latest electrode reading.
4	224	56	1.5	54.5	
1-1958	224	56	1.5	54.5	
2	224	56	1.5	54.5	
3	224	56	1.5	54.5	
4	224	56	1.5	54.5	
1-1959	224	56	1.5	545	
2	224	55	.5	54.5	
3	224	55	.5	54.5	
4	224	55	.5	54.5	

204-T-2 WHC-MR-0132

Waste Status Summary of 204-T Tank-Capacity 55,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid In Storage	Solids In Storage	Remarks
1-1960	224	54	.5	54.5	
2	224	54	.5	54.5	
3	224	54	.5	54.5	
4	224	54	.5	54.5	
1-1961	---	---	---	---	
2	224	52	2.5	54.5	6 Month Report
3	---	---	---	---	
4	224	52	2.5	54.5	6 Month Report
1-1962	---	---	---	---	
2	224	54	---	54	6 Month Report
3	---	---	---	---	
4	224	52	2	50	6 Month Report Latest electrode reading.
1-1963	---	---	---	---	
2	224	52	2	50	6 Month Report New electrode reading.
3	---	---	---	---	
4	224	52	2	50	6 Month Report
1-1964	---	---	---	---	
2	224	52	2	50	6 Month Report
3	---	---	---	---	
4	224	52	2	50	6 Month Report
1-1965	---	---	---	---	
2	224	52	8	44	6 Month Report
3	224	52	8	44	
4	224	52	8	44	
1-1966	224	52	8	44	
2	224	52	8	44	
3	224	52	8	44	
4	224	52	8	44	
1-1967	224	52	8	44	
2	224	52	8	44	
3	224	52	8	44	
4	224	52	8	44	
1-1968	224	52	8	44	
2	224	51	7	44	
3	224	51	7	44	
4	224	51	7	44	
1-1969	224	51	7	44	
2	224	51	7	44	
3	224	51	7	44	
4	224	51	7	44	

204-T-3 WHC-MR-0132

Waste Status Summary of 204-T Tank-Capacity 55,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid In Storage	Solids In Storage	Remarks
1-1970	224	51	7	44	
2	224	51	7	44	
3	224	51	7	44	
4	224	51	7	44	
1-1971	224	51	7	44	
2	224	51	7	44	
3	224	51	7	44	
4	224	51	7	44	
1-1972	224	51	7	44	
2	224	51	7	44	
3	224	51	7	44	
4	224	51	7	44	
1-1973	224	51	7	44	
2	224	50	6	44	
3	224	50	6	44	
4	224	50	6	44	
1-1974	224	50	6	44	
2	224	50	6	44	
3	224	50	6	44	
4	224	50	6	44	
1-1975	224	50	6	44	
2	224	50	6	44	
3	224	50	6	44	
4	224	50	6	44	
1-1976	224	49	5	44	Removed from service 1 to 101-T.
2	---	44	0	44	Removed from service 5 to 101-T.
3	---	44	0	44	Inactive salt well pumping.
4	---	44	0	44	
1-1977	---	44	0	44	Salt well, pump
2	---	44	0	44	
3	---	44	0	44	Inactive current
4	---	44	0	44	Inactive current

204-T-4 WHC-MR-0132

Waste Status Summary of 204-T Tank-Capacity 55,000 Gallons

Qtr.-Year	Type Waste	Total Vol.	Liquid in Storage	Solids in Storage	Remarks
1-1978	-	37	0	37	Prim. Stab. New Solids Level 1-31-78
2-	-	37	0	37	
3-	-	37	0	37	
4-	-	37	0	37	
1-1979	-	37	0	37	
2-	-	37	0	37	
3-	-	37	0	37	
4-	-	37	0	37	
1-1980	-	37	0	37	New Photo 3-18-80
2-	-	37	0	37	
3-	-	38	0	38	
4-	-	38	0	38	

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ORIGIN OF WASTE IN SINGLE-SHELL TANK 241-T-104

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Abstract: A review of waste transfer documentation was conducted to determine the origin of waste transferred into single-shell tank 241-T-104. This review was conducted to support decisions concerning disposition of the waste present in this tank.

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(2) Title
Origin of Waste in Single-Shell Tank 241-T-104

Change Control Record

(3) Revision	(4) Description of Change - Replace, Add, and Delete Pages	Authorized for Release	
		(5) Resp. Engr. (print/sign/date)	(6) Resp. Mrg. (print/sign/date)
R1	Add - Executive Summary and Section 4: Discussion that the radionuclide inventory is based on analyses of two core samples and used 10-11-2004 best basis inventory. Add - Section 2: Expanded discussion on types of records reviewed and information available in each of these records. Add - Sections 2.2 & 3.1: Added discussion that plutonium precipitate separated from uranium and fission products was washed three times and the wash water combined with the uranium and fission product solution. Add - Section 2.2.5: Discussed that 1C/CW waste was transferred to tank 241-T-104 from February through September 1954. Change - Section 2.2.5: Date that 1C/CW waste was transferred to tanks 241-TY-101 and 241-TY-103 from November 1954 to October 20, 1954. Add - Section 3.1: Spent nuclear fuel reprocessing completed in the 221 BiPO4 process when plutonium was separated from the metal waste. Change - Table 3: Sr concentrations reported for samples from tanks T-108 (top), T-108 (bottom), C-112 and average for 1C/CW were incorrectly transcribed from the reference document. Error was corrected. Add - Section 3.1.1: Included discussion on off-gas scrubbers and silver chemical reactors that were installed in the 221 BiPO4 Plants.	M.E. Johnson 10/11/04 Michael E. Johnson	S Mackey 12/15/04 STEWART MACKEY

RPP-16129
Revision 1

ORIGIN OF WASTE IN SINGLE-SHELL TANK 241-T-104

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EXECUTIVE SUMMARY

A review of waste transfer documentation was conducted to determine the origin of waste transferred into single-shell tank 241-T-104. This review was conducted to support decisions concerning disposition of the waste present in this tank.

Tank 241-T-104 was used to periodically receive first decontamination cycle (1C) waste and coating removal waste (CW) from the 221-T Bismuth Phosphate Plant from March 11, 1946, through October 19, 1954. No other waste types were received and stored in tank 241-T-104.

In the bismuth phosphate process, irradiated nuclear fuel elements were processed to recover plutonium. The aluminum coating on the fuel elements was first dissolved and separated from the uranium fuel elements. Then, the uranium fuel elements were dissolved. The plutonium was separated from the uranium and the majority of the fission products by carrier precipitation using bismuth phosphate. The uranium and fission product waste (so-called metal waste) was neutralized and transferred to various single-shell tanks. The plutonium and bismuth precipitate were processed through two additional precipitation steps to separate phosphate insoluble fission products (e.g., cerium, niobium, ruthenium, and zirconium) from the plutonium. These two precipitation steps were known as the first decontamination cycle (1C) and second decontamination cycle (2C). The coating removal waste (CW) was transferred with the 1C waste to various single-shell tanks. Similarly, the 2C waste was also transferred to various single-shell tanks.

During storage in the single-shell tanks, the 1C/CW waste precipitated solids which contained primarily bismuth, plutonium, americium, uranium, sodium, phosphate, sulfate, and metals. The 1C/CW supernatant contained primarily aluminum, sodium, nitrate, and cesium-137. As a result, tank 241-T-104 contained settled 1C/CW solids (i.e., bismuth and plutonium precipitate) and 1C/CW supernatant. The 1C/CW supernatant was removed from tank 241-T-104 and processed in the 242-T Evaporator (April through July 1951) or disposed in the east section of trench 216-T-14 (January 14, 1954). The interstitial liquid was removed from the 1C/CW sludge present in tank 241-T-104 and transferred to other underground storage tanks in two campaigns conducted February 1976 to August 1977 and March 24, 1996 to May 30, 1999.

Core samples of the sludge stored in tank 241-T-104 were obtained in 1992 and analyzed to determine chemical and radiochemical constituent concentrations. The concentration of transuranic elements present in the 1C/CW sludge contained in tank 241-T-104 is approximately 159.8 ηCi/g, based on the analytical results of the core samples for Pu-238, Pu-239, Pu-240 and Am-241. The average concentrations of cesium-137 and strontium-90 in the 1C/CW sludge as analyzed in the tank 241-T-104 core samples are approximately 0.155 μCi/g and 2.03 μCi/g, decay corrected to January 1, 2004.

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LIST OF TERMS

1C	first cycle of the bismuth phosphate plutonium decontamination process
2C	second cycle of the bismuth phosphate plutonium decontamination process
5-6	low activity cell drainage waste
cc	cubic centimeters
Ci	Curies
CW	Coating waste
DOE	U.S. Department of Energy
HLW	high-level waste
kg	kilograms
kL	kiloliters
LLW	low-level waste
MW	Metal waste
TRU	transuranic
ηCi/g	nanocuries per gram
μCi/cc	microcuries per cubic centimeters
μCi/g	microcuries per gram
μg/cc	micrograms per cubic centimeters
μg/g	micrograms per gram

1.0 INTRODUCTION

The origin of the waste in tank 241-T-104 has been reviewed to provide information for determining the disposition of this waste. Section 2.0 discusses the origin of waste transferred into and removed from single-shell tank 241-T-104. Section 3.0 provides a description of the different types of wastes that were generated at the Hanford Site chemical processing plants and transferred to single-shell tank 241-T-104. Section 4.0 provides a discussion on the radionuclide analyses of the waste in single-shell tank 241-T-104. Section 5 summarizes the waste types that were transferred into single-shell tank 241-T-104.

2.0 WASTE TRANSFER INTO AND WASTE REMOVAL FROM TANK 241-T-104

This section provides a brief description of single-shell tank 241-T-104 and summarizes waste transfers into and waste removal from these tanks. In order to determine the origins of the waste presently stored in single-shell tank 241-T-104, publicly available reports for the Hanford Site were reviewed. Documents reviewed included the Hanford site contractors' monthly reports (1945 through 1975), Army Corp of Engineers monthly reports (December 1944 through December 1946), U. S. Atomic Energy Commission monthly reports (1947 through 1954), waste disposal reports (1948 through 1975), tank farm waste status summary reports, and miscellaneous letters and technical reports.

The Hanford site contractors' monthly reports for January 1945 through July 1951 list the volume of waste stored in the single-shell tanks, with the exception of the B-200 and T-200 series single-shell tanks. No records were located that provided the volume of wastes stored in the single-shell tanks from August 1951 through February 1952. Beginning in March 1952, waste transfers and the volume of waste stored in each single-shell tank were reported for each tank in a waste status summary report.

With the exception of the waste status summary reports, all reports cited in this section are available electronically from the Hanford Declassified Document Retrieval System at http://www2.hanford.gov/declass/ or the U.S. Department of Energy (DOE) Information Bridge at http://www.osti.gov/bridge/. The waste status summary reports are available only as photocopies from Hanford Site Records Information Management Services organization.

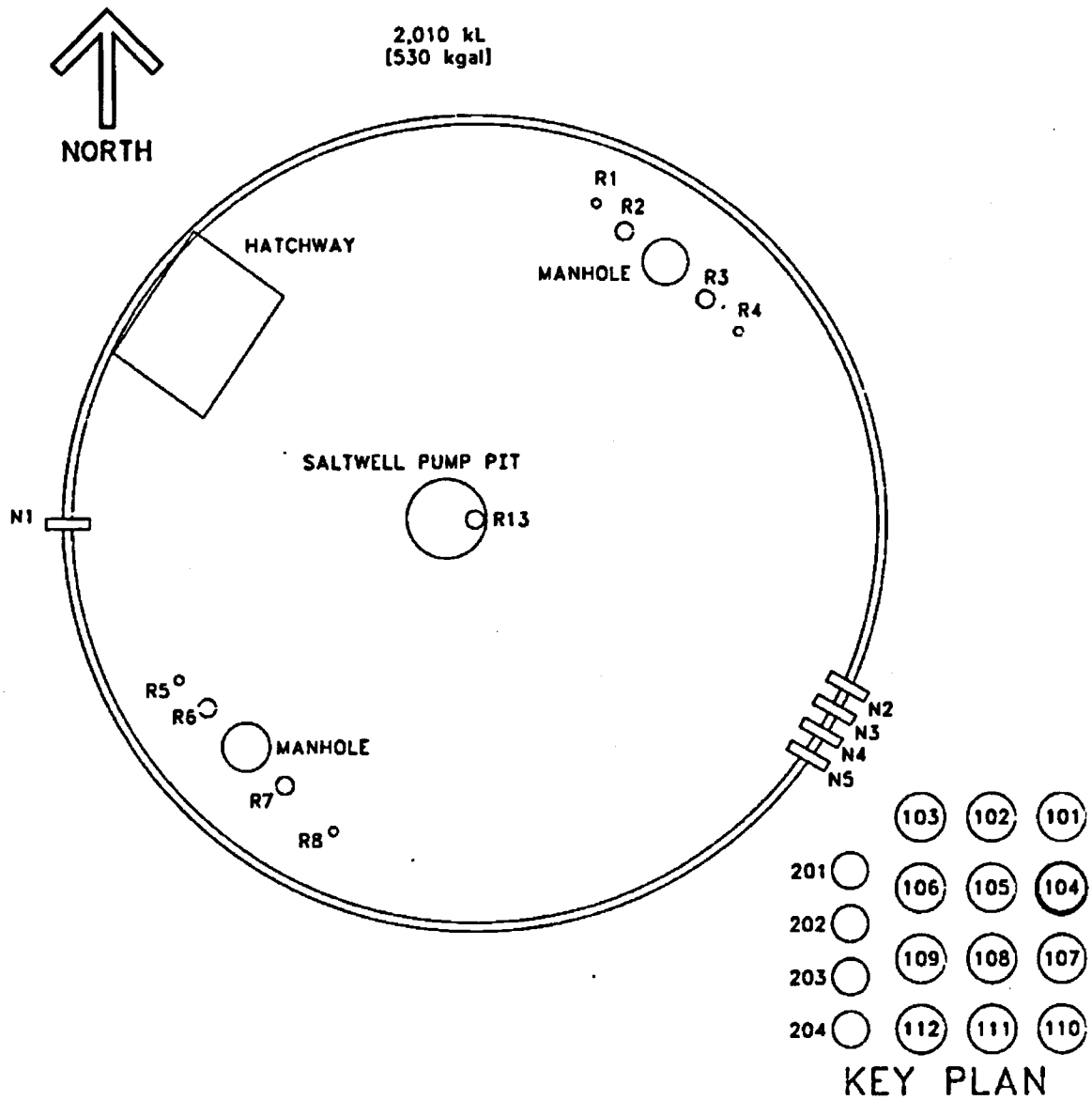
2.1 DESCRIPTION OF TANK 241-T-104

Single-shell tank 241-T-104 was originally constructed in 1944 as part of the Manhattan Project (HW-10475-C, chapter IX) and is one of the twelve, 100-series tanks in 241-T Tank Farm. Figure 1 provides a plan view of tank 241-T-104. The 100-series tanks are seventy-five-foot diameter underground tanks made of reinforced concrete with a steel liner on the bottom and sides. The steel liner extends to a height of nineteen-foot. Each 100-series tank has a design capacity of 530,000 gallons at a liquid depth of sixteen-feet, eight-inches. The 241-T Tank Farm also includes four 200-series tanks that are of similar construction as the 100-series tanks, but are only twenty-foot diameter, and each has a capacity of 55,000-gallons.

Single-shell tank 241-T-104 has five nozzles identified as N1 through N5. Tank 241-T-105 was connected via an underground overflow pipeline (nozzle N1 in Figure 1) to allow waste to cascade to tank 241-T-105. Tank 241-T-105 was also connected via a separate underground overflow pipeline to tank 241-T-106, which allowed waste to cascade from tank 241-T-104 into tank 241-T-105 and then into tank 241-T-106. In addition to the overflow piping, each tank is equipped with four, 3-inch diameter stainless steel inlet pipes.

Originally, three of the four inlet pipes (nozzles N2 through N4 in Figure 1) on tank 241-T-104 were connected to diversion box 241-T-153. The four inlet pipes for tanks 241-T-105 and 241-T-106 were blanked off close to each tank (HW-10475-C, page 907 and 908). However, on July 23, 1946, piping modifications were conducted to allow the direct transfer of waste from 221-T Plant through diversion box 241-T-153 to tank 241-T-105 (H-2-578, H-2-755, and HAN-45762, pages 26 and 32). This allowed tank 241-T-105 to be independently filled with waste that could then cascade through the overflow pipeline to tank 241-T-106. Waste transfers into tank 241-T-104 and the operation of the tanks 241-T-104, 241-T-105 and 241-T-106 as a cascade are discussed in more detail in Section 2.2.

Figure 1. Tank 241-T-104 Plan View



2.2 WASTE TRANSFERS FOR TANK 241-T-104

The following sections describe in chronological order the waste transferred into tank 241-T-104 and the waste disposition. Appendix A provides a listing of the waste volume present in tank 241-T-104 from 1945 through 1977. The only types of waste transferred into tank 241-T-104 were coating removal waste and first decontamination cycle waste from the bismuth phosphate process conducted in the 221-T Plant.

In the bismuth phosphate process, irradiated nuclear fuel was processed to separate plutonium using the bismuth phosphate process. The coating on the irradiated nuclear fuel was initially dissolved and separated from the uranium fuel elements. The dissolved fuel coating was referred to as coating removal waste. The uranium fuel elements were then dissolved. Plutonium was separated from the dissolved uranium by precipitation and centrifugation of the plutonium precipitate. The plutonium precipitate was washed three times and the wash solution combined with the uranium and fission products that remained in solution. The uranium and fission products that remained in solution were discarded as waste to the single-shell tanks. The plutonium precipitate was dissolved and processed through two successive precipitation steps to separate fission products from the plutonium. These precipitation steps were referred to as the first and second decontamination cycles. The waste from each decontamination cycle was referred to as 1C and 2C waste. The plutonium was further processed in the 224-T Concentration Building and the 231-Z Isolation Plant before shipment from the Hanford Site. Section 3.0 discusses the bismuth phosphate process in greater detail.

2.2.1 1C/CW Waste Storage (March 1946 – July 1946)

Irradiated nuclear fuel was first processed in 221-T Plant beginning on December 26, 1944 (HW-7-1293-DEL, page 19). The first decontamination cycle (1C) waste was combined with the coating removal waste (CW) and transferred to the cascade of tanks 241-T-107, 241-T-108, and 241-T-109. The combined 1C/CW waste was reported as being collected in tank 241-T-107 in February 1945 (HW-7-1338-DEL, page 22). Tanks 241-T-107, 241-T-108, and 241-T-109 continued to receive the combined 1C/CW waste until March 10, 1946, when these tanks were reported as being filled (HW-7-3751-DEL, pages 20 and 21).

On March 11, 1946, a pipeline was established to transfer the combined 1C/CW waste from the 221-T Plant to tank 241-U-110 in the 241-U Tank Farm. However, this transfer line developed a plug shortly after first being used and the combined 1C/CW waste was transferred to tank 241-T-104 (HW-7-3751-DEL, pages 20 and 21). Tank 241-T-104 was filled with the combined 1C/CW waste in July 1946.(HW-7-4542-DEL, pages 21-22).

Tanks 241-T-104, 241-T-105, and 241-T-106 were originally designated as a spare set of tanks for receipt of second decontamination cycle (2C) waste from the 221-T Plant. Collection of the combined 1C/CW waste in tank 241-T-104 was considered at the time to be a temporary measure. In order to allow the collection of 2C waste from 221-T Plant in tanks 241-T-105 and

241-T-106, a separate transfer pipeline was established to the inlet of tank 241-T-105 on July 17, 1946 (H-2-578 and HAN-45762, pages 27 and 32). Tank 241-T-105 was used to store 2C waste from the 221-T Plant from July 23, 1946 (HW-7-4542-DEL, page 22) through April 1948, after which the 2C supernatant waste was discharged to a crib (HW-9922-DEL, page 31). Tank 241-T-106 began to receive 2C waste through the overflow line from tank 241-T-105 in June 1947 (HW-7-7454-DEL, page 26) and was filled in March 1948 (HW-9595-DEL, page 32). The 2C supernatant waste contained in tank 241-T-106 was discharged to a crib from July 1948 (HW-10714-DEL, page 32) through August 3, 1948 (HW-10993-DEL, page 35).

The plug in the transfer line from 221-T Plant to the 241-U Tank Farm was successfully removed in April 1946 (HW-7-4004-DEL, page 20). The combined 1C/CW waste from 221-T Plant was diverted to the cascade of tanks 241-U-110, 241-U-111, and 241-U-112 on July 22, 1946 (HW-7-4542-DEL, page 21-22). The cascade of tanks 241-U-110, 241-U-111, and 241-U-112 were filled with the 1C/CW waste from the 221-T Plant in May 1948 (HW-10166-DEL, page 33).

After emptying the 2C supernatant waste from tank 241-T-105, the combined 1C/CW waste was transferred from 221-T Plant to tank 241-T-105 beginning in May 1948 (HAN-45807-DEL, page 55). Waste began to cascade from tank 241-T-105 into tank 241-T-106 in August 1948. Tank 241-T-105 continued to receive 1C/CW waste through January 1949, at which tanks 241-T-104, 241-T-105, and 241-T-106 were all filled with 1C/CW waste (HW-12391-DEL, page 38). The 1C/CW waste generated at the 221-T Plant was then transferred to the cascade of tanks 241-TX-109 through 241-TX-112 (HW-12391-DEL, page 38).

2.2.2 1C/CW Supernatant Evaporation (March 1951 – July 1951)

The pH of the 1C/CW waste was adjusted to approximately pH 7 in the 221-T Plant before transfer to the single-shell tanks. This pH adjustment was conducted to cause the precipitation of bismuth and plutonium in the 1C/CW waste so that the supernatant would contain a lower concentration of plutonium (HW-7-2706-DEL, page 21). As a result, tank 241-T-104 contained settled 1C/CW solids (i.e., bismuth and plutonium precipitate) and 1C/CW supernatant (HW-20991-DEL, page 53).

The 1C/CW supernatant contained in tank 241-T-104, along with that contained in the other tanks in 241-T Farm were transferred to tanks 241-TX-117 and 241-TX-118 from March 1951 (HW-20671-Del, page 56) through July 1951 (HW-21802-DEL, page 42). The 1C/CW supernatant was transferred from tanks 241-TX-117 and 241-TX-118 to the 242-T Evaporator for evaporation. Processing of the 1C/CW supernatant from the 241-T Farm tanks in the 242-T Evaporator was conducted from April 28, 1951 (HW-20991-DEL page 54 and HAN-63671, page 40) through July 1951 (HW-21802-DEL, page 42). The concentrated 1C/CW supernatant waste (i.e., evaporator bottoms) was stored in tanks 241-TX-116 and 241-TX-117. The evaporator bottoms in tanks 241-TX-116 and 241-TX-117 were eventually processed again through the 242-T Evaporator to further concentrate these wastes for storage in tanks 241-TX-110 and 241-TX-111.

2.2.3 1C/CW Waste Cascade Filling (August 1951 – December 1951)

After evaporating the 1C/CW supernatant, tank 241-T-104 was again used to store 1C/CW waste from the bismuth phosphate process conducted in 221-T Plant. Tank 241-T-104 was operated as a cascade with tanks 241-T-105 and 241-T-106. Tanks 241-T-104, 241-T-105, and 241-T-106 were reported as being filled with 1C/CW waste in August 1951, October 26, 1951, and December 22, 1951, respectively (HW-33591, page 12).

2.2.4 Trench Disposal of 1C/CW Supernatant (January 1954)

Plans were made to allow the 1C/CW waste to remain in the cascade of tanks 241-T-104, 241-T-105, and 241-T-106 for one-year to allow for the decay of short-lived fission products, after which the supernatant was to be processed in the 242-T Evaporator (HW-27838, page 32). However, evaporation of the supernatant contained in these tanks was not conducted.

Instead, the 1C/CW supernatant contained in tank 241-T-104 was discharged to the east section of trench 241-T-1 (later renamed to trench 216-T-14) on January 14, 1954 (HW-33591, page 12). The 1C/CW supernatant contained in tanks 241-BX-110, 241-BX-111, 241-BX-112, 241-BY-106, 241-BY-110, 241-T-105, 241-T-106, 241-TX-109, 241-TX-110, and 241-TX-111 and 1C/CW evaporator bottoms contained in tanks 241-B-107, 241-B-108, 241-B-109, 241-TY-101 and 241-TY-102 were also discharged to trenches from January 1954 through November 1954 (HW-33591, pages 11 and 12 and HW-38562, pages 10, 28 and 29). The disposal of 1C/CW supernatant to these trenches was based on the concept of retaining fission products, plutonium, and uranium in the soil column. Trench disposal of the 1C/CW supernatant and evaporator bottoms was thought to be an economical method for providing additional capacity in the single-shell tanks for storage of wastes with higher radioactivity (HW-34281).

2.2.5 1C/CW Waste Cascade Filling (February 1954 – October 1954)

Beginning on February 23, 1954, 1C/CW waste was again transferred to tank 241-T-104 (HW-31126, page 5). The tank was filled to approximately 6-inches above the cascade overflow line and waste began to cascade to tank 241-T-105 in March 1954 and to tank 241-T-106 in June 1954. The cascade of tanks 241-T-104, 241-T-105 and 241-T-106 was filled by the end of September 1954 (HW-33396, page 5) and did not receive any additional 1C/CW waste from 221-T Plant.

On October 20, 1954, modifications to the bismuth phosphate process were conducted in 221-T Plant to segregate the coating removal waste (CW) from the first decontamination cycle (1C) waste (HW-33585-DEL, page Ed-8 and HW-33544, page 5). The coating removal waste was transferred directly to tank 241-T-105, which cascaded through the overflow line into tank 241-T-106. Additional storage space for the coating removal waste was provided in tank 241-T-105 by pumping some of the supernatant from tank 241-T-105 to tank 241-TX-118 for processing in the 242-T Evaporator (HW-33904, page 5).

The 1C waste was treated in 221-T Plant with potassium ferrocyanide, nickel sulfate, and sodium hydroxide to precipitate nickel ferrocyanide, which scavenged cesium-137 and strontium-90 from the supernatant (HW-33184 and HW-33499). The treated 1C waste was discharged to tanks 241-TY-101 and 241-TY-103 from October 20, 1954 (HW-33544, page 7) through March 20, 1956, when the final processing of irradiated fuels for plutonium recovery was completed in the 221-T Plant (HW-42219-DEL, page Ed-5). The nickel ferrocyanide precipitate and the scavenged fission products settled in these tanks, with the supernatant transferred in March 1955 to tank 241-TX-118 for processing in the 242-T Evaporator (HW-36001, page 7) and disposal in 216-TY trenches from October 1955 (HW-44784, page 42) through November 1956 (HW-48518, page 34).

After October 20, 1954, tank 241-T-104 was no longer used to receive waste from the 221-T Plant. No other transfers of waste into tank 241-T-104 have occurred.

2.2.6 Saltwell Pumping / Interim Stabilization

The 1C/CW waste remained undisturbed in tank 241-T-104 until July 1969, when approximately 48,000-gallons of supernatant were transferred from tank 241-T-104 to tank 241-TY-103 (ARH-1200 C, page 7).

Removal of additional liquid from tank 241-T-104 was conducted from February 27, 1976 through August 17, 1977 as part of the program to remove interstitial liquid (i.e., saltwell pumping) from the single-shell tanks (letter 60410-78-092 and DS-022676). A total of 38,200 gallons of liquid waste was reported as being pumped from tank 241-T-104 to tank 241-T-101 during this period. In May 1978, saltwell pumping of liquids from tank 241-T-104 was attempted again. However, the pump was reported as inoperable. Saltwell pumping of liquids from tank 241-T-104 was resumed on September 11, 1978 and concluded on December 26, 1978 (DS-022676). An additional 7,420 gallons of liquid waste were transferred from tank 241-T-104 to tank 241-T-101. The height of waste in tank 241-T-104 was reported as 13-feet and 1.25-inches following saltwell pumping. The volume of waste in tank 241-T-104 was approximately 412,000 gallons based on the waste height measurement.

Interim stabilization of tank 241-T-104 was conducted from March 24, 1996 through May 30, 1999 (HNF-EP-0182 revision 172, page B-15). Approximately 150,000 gallons of liquid were pumped from tank 241-T-104 to the double-shell tank system, leaving an estimated 316,800 gallons of sludge in this tank (HNF-SD-RE-TI-178, page 200). Tank 241-T-104 was declared having been interim stabilized on November 19, 1999 (HNF-EP-0182 revision 172, page B-13).

2.2.7 Comparison with Other Reports

Waste transfers into and waste removals from tank 241-T-104 are summarized in *A History of the 200 Area Tank Farms* (WHC-MR-0132) for 1945 through 1980, *Historical Tank Content Estimate for the Northwest Quadrant of the Hanford 200 West Area* (HNF-SD-WM-ER-351), and *Waste Status and Transaction Record Summary (WSTRS) Rev. 4* (LA-UR-97-311). The information cited in Sections 2.2.1 through 2.2.10 is in agreement with these previous reports. These previous reports accurately state the volume of waste transferred into and removed from tank 241-T-104, as well as the volume of solids and total waste stored.

3.0 TYPES OF TANK WASTE GENERATED AT THE HANFORD SITE CHEMICAL PROCESSING PLANTS

There were numerous irradiated nuclear fuel reprocessing, research and development, plutonium processing, and waste management activities conducted at the Hanford Site starting in 1944. These irradiated nuclear fuel reprocessing, research and development, plutonium processing, and waste management activities conducted in the processing plants are discussed further in the DOE/RL-97-02, *National Register of Historic Places Multiple Property Document Form - Historic, Archaeological and Traditional Cultural Properties of the Hanford Site, Washington February 1997.*

It has been established in Section 2.0 that first decontamination cycle (1C) waste mixed with coating removal waste (CW) from the 221-T Bismuth Phosphate plant was transferred into tank 241-T-104. The following sections provide a discussion of the wastes originating from the bismuth phosphate plant operations.

3.1 B AND T BISMUTH PHOSPHATE PROCESS PLANTS

B- and T-Plants were constructed in 1944 through 1945 to separate plutonium from irradiated nuclear fuel using the bismuth phosphate process. Figure 2 shows a summary of the 221-B/T Plant bismuth phosphate process, which is referred to throughout this discussion. The bismuth phosphate process was operated in B-Plant from April 1945 (HW-7-1649-DEL, page 21) through June 1952 (HW-25227-DEL, pages Ed-5 and Ed-6), after which the inventory of radioactive materials was removed from the facility from July 1952 through March 1953 (HW-27774). The bismuth phosphate process was operated in T-Plant from December 1944 (HAN-45800-DEL, page 4) through March 1956, after which the inventory of radioactive materials was removed from the facility from March 1956 (HW-42219-DEL, page ED-5) through September 1956 (HW-45707-DEL, page D-5). T-Plant was placed in layaway status in October 1956 (HW-46432-DEL, page D-5). T-Plant was re-activated in 1960 and is currently in use for equipment decontamination and storage of commercial irradiated nuclear fuel elements.

In the bismuth phosphate process, the aluminum cladding of spent nuclear fuel elements was dissolved in boiling sodium nitrate solution, to which sodium hydroxide was slowly added (HW-10475-C, page 403). The cladding removal waste sometimes referred to as coating waste (CW) was transferred to single-shell underground storage tanks (see item [1] in Figure 2).

Reprocessing of the spent nuclear fuel commenced with the dissolution of the uranium fuel elements. The uranium fuel elements (see item [2] in Figure 2) were dissolved in nitric acid (HW-10475-C, chapter IV, page 405). Water and sulfuric acid were added to the dissolved uranium metal solution and the mixture was then transferred to the plutonium extraction section. The sulfuric acid formed a uranyl sulfate complex that prevented uranium precipitation as a phosphate in the subsequent plutonium extraction step (HW-10475-C, page 418).

Plutonium was extracted from the acid solution by addition of bismuth nitrate and phosphoric acid to form a bismuth phosphate carrier precipitate (HW-10475-C, page 503). The plutonium

and bismuth phosphate carrier precipitate was centrifuged and washed three times with water to separate the acidic supernatant from the plutonium precipitate (see item [3] in Figure 2). The acidic solution remaining after the plutonium precipitation contained about 99 percent of the uranium, about 90 percent of the fission products. This separation process also removed and reduced the gamma radiation activity level in the plutonium precipitate by a factor of 10. However, zirconium is phosphate insoluble and zirconium-95 (10 percent of the activity) stayed with the plutonium product. The acidic uranium solution was then neutralized and transferred to the underground single-shell tanks as metal waste (MW). Recent laboratory testing of the bismuth phosphate flowsheet confirms this partitioning of radionuclides (internal letter 7G300-02-NWK-024, "Bismuth Phosphate Process Radionuclide Partition Factors for the Hanford Defined Waste Model"). The laboratory tests indicate the percentage of cesium-137 and strontium-90 partitioned to the metal waste may have been as high as 100 percent and 89 percent, respectively.

After separating and washing the plutonium precipitate from the metal waste, reprocessing of spent nuclear fuel was completed in the 221 Plant Bismuth Phosphate process. Plutonium decontamination was conducted in the remainder of the 221 Plant Bismuth Phosphate process. The plutonium-bearing cake was dissolved in nitric acid and further decontamination of the plutonium to separate fission products was conducted (HW-10475-C, chapter VI). Sodium bismuthate, sodium dichromate, or potassium permanganate was added to oxidize the plutonium to the +6 valence-state. This step caused the bismuth phosphate to precipitate phosphate insoluble fission products (e.g., cerium, niobium, ruthenium, and zirconium), leaving the plutonium in solution. The precipitate was separated from the plutonium-bearing solution using centrifuges and washed to remove soluble plutonium. The plutonium was reduced to the +4 valence state to form a precipitate that could be separated from the remaining soluble fission products by centrifugation.

The fission products separated from the plutonium product during this first cycle of the decontamination process (designated as 1C waste) were transferred to the single-shell tanks. The 1C waste (see item [4] in Figure 2), contained approximately 10 percent of all fission products and approximately 1.4 percent of the plutonium present in the original fuel charged to the plant (HW-23043, pages 20 and 22). After 1951, the bismuth phosphate process flowsheet was modified to include cerium and zirconium scavenger precipitation in the 1C by-product step to remove lanthanide and zirconium radionuclides from the plutonium product (HW-23043, page 16).

The plutonium solids from the first decontamination cycle were again dissolved in nitric acid. A second decontamination cycle (see item [5] in Figure 2) was conducted to reduce the gamma activity level by a factor of 10,000 from that in the previous dissolved metal solution, giving an overall process decontamination factor of 100,000 below that of the original solution (HW-10475-C, page 627). The second decontamination step essentially repeated the steps previously described for the first cycle decontamination. The plutonium product from the bismuth phosphate process was subsequently concentrated in the 224-T and 224-B buildings using a lanthanum fluoride precipitation process.

The second decontamination cycle wastes (designated as 2C) were also transferred to the

single-shell tanks. The 2C waste contained less than 0.1 percent of the uranium and fission products and about 0.4 percent of the plutonium present in the original fuel charged to the plant (HW-23043, pages 26 and 28).

During operation of B-Plant, the 1C waste was combined with the coating removal waste and transferred to the same single-shell tank. This same practice was conducted in T-Plant from December 1944 through October 19, 1954. Beginning on October 20, 1954, nickel ferrocyanide scavenging of the 1C waste was conducted in T-Plant to precipitate cesium-137 and strontium-90 (HW-33585-DEL, page Ed-8, and HW-33184). The precipitated 1C waste slurry was transferred separate from the coating removal waste to separate single-shell tanks for settling of the precipitate and discharge of the scavenged (i.e., cesium and strontium depleted) supernatant to a crib.

Table 1 provides the flowsheet estimated compositions of the neutralized CW, MW, 1C, and 2C waste solutions generated from the 221-B/T bismuth phosphate plants based on the October 1, 1951 flowsheet (HW-23043). Additional analyses of the supernatant fraction of MW, 1C/CW, and 2C that was stored in single-shell tanks are provided in Tables 2 and 3.

These sample analyses support that the 2C waste contained less than 0.1 percent of the fission products. Analyses of the combined 2C / 224 building / tank 5-6 waste supernatant stored in tank 241-T-112 conducted on August 6, 1952 and September 24, 1952 indicate that the total beta emitters was comprised of 35 to 50 percent ruthenium, 35 to 50 percent cesium, 4 to 8 percent cerium, yttrium, and other rare earths, and 6 to 11 percent undetermined (HW-27035, page 8).

Figure 2. Bismuth Phosphate Process Diagram

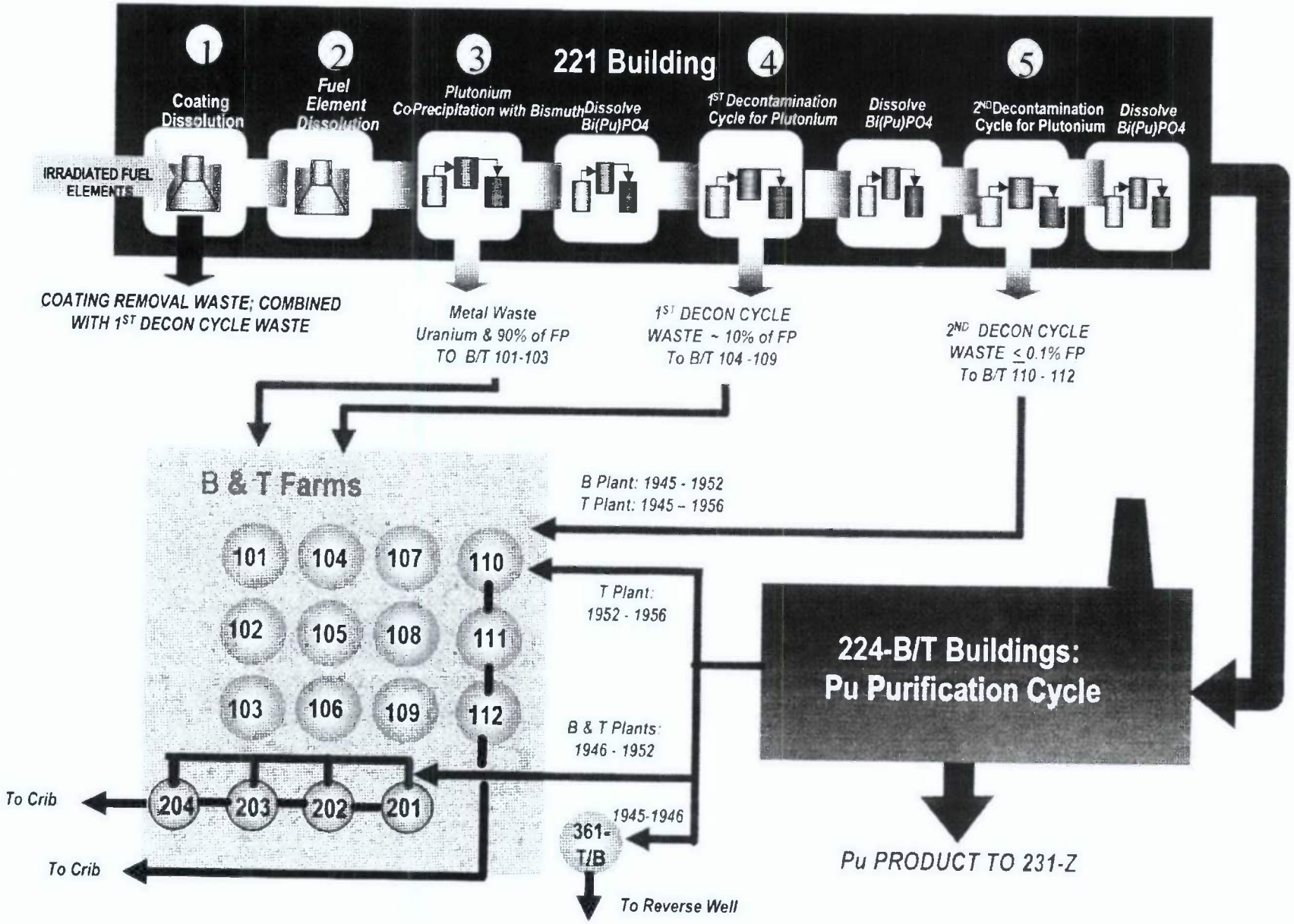


Table 1. Estimated Composition of Bismuth Phosphate Plant Wastes From October 1, 1951 Flowsheet [1]

Analyte [2]	Coating Removal Waste	Metal Waste	First Decontamination Cycle (1C) Waste	Second Decontamination Cycle (2C) Waste	224 Building Waste
Plutonium	3.3E-04	2.0E-04	6.0E-07 [4]	1.6E-07 [5]	1.68E-04 [6]
Uranium	0.15		0.235 [4]	Not reported	2.04E-05
Gamma	6.6E+04	1.3E+07	2.3E+06 [4]	1.13E+04 [5]	1.13E+02 [6]
Sodium Aluminate ($NaAlO_2$)	95.1				
Sodium Hydroxide (NaOH)	43.6				
Sodium Nitrate ($NaNO_3$)	61.8				
Sodium Nitrite ($NaNO_2$)	56.0				
Sodium Silicate ($NaSiO_3$)	4.3				
Uranyl nitrate (UHN) [3]		132			
Fluorine (F)					5.6
Nitrate (NO_3)		9.7	93.1	61.3	42.4
Sulfate (SO_4)		24.4	4.73	3.61	0.35
Phosphate (PO_4)		25.2	26.2	23.0	3.05
Sodium (Na)		83.2	47.3	36.7	36.8
Bismuth (Bi)			2.59	1.31	1.18
Cerium (Ce)			0.030		
Lanthanum (La)					0.49
Manganese (Mn)					0.33
Zirconium (Zr)			0.030		
Iron (Fe)			1.37	1.82	
Chrome (Cr)			0.16	0.06	0.17
Ammonia (NH_4)			1.98	1.71	0.12
Silicon Hexa-Fluoride (SiF_6)			4.35	3.67	
Volume per Batch (gallons)	795	2,380	2,040	2,090	2,200

Notes:

[1] See HW-23043.

[2] Analyses are reported in grams per liter, except for gamma activity, which is counts per minute per mL.

[3] HW-23043, page 31, notes that uranium is not actually present in this form, but is probably as $NaUO_2PO_4$ and $Na_4(UO_2)_2CO_3$.

[4] Pu and Gamma concentrations were calculated from the compositions of tanks 13-4 and 14-3 (HW-23043, pages 20 and 22).

[5] Pu and Gamma concentrations were calculated from the compositions of tanks 18-4 and 19-3 (HW-23043, pages 26 and 28).

[6] Pu and Gamma concentrations were calculated from the compositions of tanks A-4, D-4, B-3, and F-8 (HW-23043, pages 39, 44, 48, and 54).

Table 2. Analyses of Bismuth Phosphate Process Supernatants Stored [1,2].

Waste Type	Tank	pH	Pu µg per Liter	Gross Beta Millicuries per Liter	Gross Gamma Millicuries per Liter	Date Sampled
Metal Waste	T-101	10.1	70	200[5]	70[5]	12-12-1946
Metal Waste	T-101	10	35	110[5]	25[5]	7-01-1947
Metal Waste	T-102	9.9	60	120	20	7-01-1947
Metal Waste	T-103	9.8	60	150	20	7-01-1947
1C/CW	B-109	9.9	.40	0.65	0.28	3-18-1947
1C/CW	C-112	9.9	12	12	4.4	3-18-1947
2C	B-111	6.9	7.2E-02	2.0E-03	3.0E-03	7-1-1947
2C	B-112	6.8	4.32E?? [3]	1.5E-03	3.0E-03	7-1-1947
Waste Type	**Tank**	**pH**	**Pu µg per Liter**	**Gross Beta Counts per minute per cc**	**Gross Gamma Counts per minute per cc**	**Date Sampled**
2C	T-110	Not reported [4]	15	4.9E+04	30	7-13-1945
2C	T-110	9.8[4]	19	6.9E+04	55	7-25-1945
2C	B-110	9.6[4]	8.5	7.0E+04	55	7-25-1945

Notes:

(1) See HW-10728 and HW-3-3220.

(2) Solids formed in each of wastes, settling to the bottom of each tanks. These sample analyses are for the supernatant only and are not representative of the sludges.

(3) The reported Pu sample analyses for tank B-112 seem to be in error and lacking an exponent in HW-10728.

(4) Prior to October 1945, the 1C and 2C wastes were neutralized to a pH of approximately 10. The waste collected in tanks 241-B-110, 241-B-111, 241-B-112, 241-T-110, 241-T-111, and 241-T-112 were neutralized to about pH 7 after October 1945 to precipitate bismuth and plutonium (HW-3-3220, page 13).

(5) Decreases in gross beta and gross gamma concentrations shown for the T-101 waste samples are due to decay of fission products with short half-lives.

Table 3. Analyses of Metal Waste and First Decontamination Cycle / Coating Waste Supernatant.

Tank	Date Filled	Pu µg/cc	Gross Beta µCi/cc	Gross Gamma µCi/cc	Sr µCi/cc	Cs µCi/cc	Ru µCi/cc	Rare Earths + Y − Ce µCi/cc	Ce µCi/cc	Nb µCi/cc	Zr µCi/cc	Te µCi/cc
Analyses of Metal Waste Supernatant Following Uranium Extraction [1]												
C-106	Not specified				0.44	54.2						
BX-108	Not specified				0.26	132.4						
BX-109	Not specified				1.08	56.3						
C-112	Not specified				1.20	25 8						
C-109	Not specified				0.46	40.7						
C-111	Not specified				0.10	34.5						
Average Concentrations for Metal Waste					0.59	57.3						
Analyses of First Decontamination Cycle (1C) Waste Mixed with Coating Removal Waste (CW) [2]												
B-107	8-1945	1.7E-02	0.135	0 055	0 011	0.10						
T-107	9-1945	1.5E-03	0.170	0.093	0.0013	0.20						
B-108	12-1945	2.0E-02	0.183	0 044	0.022	0.12						
T-108 (Top)	12-1945	2.0E-02	0.25	0 073	0 012	0.17	0 0066	0 047	0.007	0.0018	0	1.2E-05
T-108 (Bottom)	12-1945	2.0E-02	0.25	0.070	0.012	Not reported	0.0065	0.029	0.0066	0 0024	0	3E-05
T-109	3-1946	2 6E-03	0.14	0 082	0 00038	0.15						
B-109	4-1946	1.8E-02	0.16	0.051	0 01	0 11						
T-104 (Top)	7-1946	3E-03	0.51	0.130	0.00013	0.13	0.058	0.004	0.051	0 028	0 010	2.4E-05
T-104 (Bottom)	7-1946	3E-03	0.52	0.160	0.00037	Not reported	0 059	0.003	0.050	0.028	0.015	3.6E-05
C-110	8-1946	2E-03	0.14	0 0067	0.00026	0.11						
C-111	11-1946	4.2E-03	0.16	0 069	0 01	0 13						
C-112	4-1947	3.1E-03	0.14	0.064	0 006	0.13						
U-110	4-1947	2.1E-04	0.13	0 069	0.00011	0.17						
U-111	10-1947	3.4E-04	0 12	0 060	0 00023	0.14						
TX-109 [3]	9-1949	2.7E-05	2 8	2.2	0 00087	0.27	0.34	0 0085	0 0035	0.34	1.2	8E-05
Average Concentrations for 1C / CW		7.67E-03	0.39	0.22	0.0058	0.15						

Notes:

[1] HW-36717, 1955, *Decontamination of Uranium Recovery Process Stored Wastes Interim Report*, General Electric Company, Richland, Washington.
[2] HW-20195, 1951, *Radioactive Content of Stored Bismuth Phosphate First Cycle Waste Supernatants*, General Electric Company, Richland, Washington.
[3] Tank TX-109 exhibits higher gross beta and gross gamma radioactivity since this tank was sampled shortly after filling and the short-lived fission products (e g., Ru, Nb, and Zr) had not decayed appreciably.

3.1.1 221-T and 221-B Plant Cell Drainage Waste

During the operation of the 221-B and 221-T Bismuth Phosphate plants, failure of process equipment, cooling jackets on process vessels, and piping occurred periodically, resulting in the discharge of cooling water, chemical solutions, and process solutions (e.g., MW, 1C, 2C wastes and plutonium product solutions) to the process cells. Each of the 40 process cells in the 221-B and 221-T Plants contained a sump that was equipped with a conductivity probe beginning in August 1946 to detect a liquid leak in the process cell (HW-7-4739-DEL, page 21). The sumps gravity drained to a 24-inch diameter vitrified clay pipe that traversed under each cell and discharged to a deep, open top, stainless steel tank, number 5-7 in section 5 (cell 10) (HW-10475-C, page 914).

Cell drainage collected in tank 5-7 was jetted to tank 5-6 or tank 5-9, which were used for sampling and chemical treatment of the cell drainage solution. Waste in tanks 5-6 and 5-9 could be jetted between these two tanks. High-activity waste collected in 221-T Plant and 221-B Plant tanks 5-9 could be jetted to single-shell tanks 241-T-107 and 241-B-107, respectively (HW-10475-C, page 918). Alternatively, the waste could be transferred to process vessels within the 221-T (or 221-B) Plant and processed to recover plutonium. An example of this practice is cited in the January 1948 monthly report for the Hanford Works (HW-8931-Del, page 28).

The T-Plant stack drainage waste was also collected as part of the cell drainage until May 28, 1951, after which the stack drainage was routed to the cascade of single-shell tanks 241-TX-113, 241-TX-114, and 241-TX-115 (HW-21260-DEL, page 58). Also, the dissolvers located in 221-B and 221-T Plant cells 5, 6 and 7 were equipped with off-gas scrubber towers in May 1948 (HAN-45807, pages 57). The dissolver off-gas scrubbers used water to adsorb iodine and remove particulates from the dissolver off-gases. The spent scrubber solution was combined with the low-activity cell drainage waste collected in tank 5-6 (HW-10728). The dissolver off-gas scrubbers were replaced with silver chemical reactors, thus eliminating the spent scrubber solution. The first silver reactor was installed in the 221-B Plant in October 24, 1950 (HW-19898 and HW-19325, page 52) and the remaining silver chemical reactors were installed in the 221-B and 221-T Plants by January 1951 (HW-20161, page 52 and HW-21826).

Waste collected in tank 5-6 was transferred to reverse well number 216-T-3 from January 1945 through August 1946. Crib number 216-T-6 was used to dispose of the cell drainage waste from August 1946 through June 1951. After June 1951, cell drainage waste was transferred to the cascade of tanks 241-T-110, 241-T-111, and 241-T-112 (HW-55176, part V). The quantity and composition of the cell drainage solutions discharged from tank 5-6 varied (see HW-20583, page 4 and HW-33591, page 25). Table 4 provides analyses of cell drainage waste that was collected in tank 5-6 and transferred to either crib 216-T-6 or to the cascade of tanks 241-T-110, 241-T-111, and 241-T-112. As evident from the analyses provided in Table 5, the neutralized, low-activity cell drainage waste contained soluble beta emitting radionuclides and plutonium.

3.1.2 221-T Equipment Decontamination Facility

In October 1958, plans were developed to convert the 221-T Plant for use as decontamination facility for equipment from the Reduction-Oxidation (REDOX) plant (HW-58051-DEL, page D-5). Work was conducted from February 1959 (HW-59434-DEL, page D-4) through June 1960 (HW-65935-DEL, page C-2) to convert the 221-T Plant. Equipment decontamination activities were initiated at the 221-T Plant in July 1960, with the receipt of a failed multipurpose dissolver from the REDOX plant (HW-66271-DEL, page C-2). Equipment decontamination waste was transferred to single-shell tanks 241-T-111 and 241-T-112 (RPP-13873, section 2.2.4). Tank 241-T-104 did not receive equipment decontamination waste.

Table 4. Composition of Tank 5-6 Cell Drainage Waste from 221-T Plant. (3 sheets)

Year	Month	Liters	Pu Grams	Total Beta Activity Curies	Comment
Tank 5-6 Cell Drainage Transferred to 216-T-6 Crib [1,2]					
1948	January	839,900	49	88	Total beta activity does not include radioactive iodine. Samples were measured for total alpha activity. Calculated Pu mass assumes that all alpha activity measured in samples was Pu. Uranium activity in samples contributed less than 8% of the total alpha activity [1].
	February	724,461	8	73	
	March	586,188	3	789	
	April	842,778	9	461	
	May	918,007	5	72	
	June	971,810	9	295	
	July	1,057,015	6	130	
	August	831,662	4	248	
	September	857,327	5	361	
	October	830,083	4	116	
	November	980,411	6	214	
No records could be located for December 1948 through August 1949.					
1949	September	260,000	32	365	
	October	360,000	41	2800	
	November	340,000	38.2	333	
	December	430,000	48	250	
1950	January	410,000	44	210	
	February	330,000	28.5	No data reported	
	March	370,000	35	No data reported	
	April	450,000	35.6	294	
	May	370,000	33.9	363	
	June	430,000	36.6	2142	
	July	520,000	43.6	600	
	August	590,000	44.9	741	
	September	480,000	42.3	850	
	October	620,000	47.3	858	
	November	540,000	50.9	600	
	December	590,000	42.1	850	
No records could be located for January 1951 through December 1951. Beginning in June 1951, Tank 5-6 cell drainage waste along with 2C waste was routed to the cascade of tanks 241-T-110, 241-T-111, and 241-T-112.					

Table 4. Composition of Tank 5-6 Cell Drainage Waste from 221-T Plant. (3 sheets)

Year	Month	Liters	Pu Grams	Total Beta Activity Curies	Comment
Tank 5-6 Cell Drainage Waste Discharged to the Cascade of Tanks 241-T-110, 241-T-111, and 241-T-112 [3,4]					
1952	January	595,000	5.2	440	
	February	498,000	6.9	850	
	March	643,000	8.2	920	
	April	623,000	8.8	660	
	May	318,000	1.8	84	
	June	392,000	3.0	97	
	July	600,000	4.1	160	Beginning in July 1952, 224 building waste, along with tank 5-6 cell drainage and 2C wastes were routed to the cascade of tanks 241-T-110, 241-T-111, and 241-T-112. Values reported are for tank 5-6 cell drainage waste only.
	August	670,000	6.5	265	
	September	260,000	1.9	675	
	October	430,000	3.0	310	
	November	490,000	2.7	95	
	December	540,000	3.3	240	
1953	January	490,000	2.4	130	
	February	530,000	3.9	480	
	March	660,000	5.0	245	
	April	390,000	2.0	180	
	May	490,000	1.8	220	
	June	660,000	3.5	590	
	July	280,000	0.9	65	
	August	490,000	2.4	100	
	September	560,000	7.8	195	
	October	560,000	6.8	1,840	
	November	710,000	8.7	1,085	
	December	740,000	8.8	885	
1954	January	830,000	10.4	1,680	
	February	820,000	14.2	16,420	
	March	860,000	18.6	5,305	
	April	540,000	8.4	2,175	
	May	790,000	10.6	1,760	
	June	810,000	9.5	2,390	
	July	1,030,000			Radionuclide content not reported separately for 5-6 Cell drainage waste from July 1954 through June 1955 (HW-38562, page 26).
	August	1,150,000			
	September	1,090,000			
	October	800,000			
	November	730,000			
	December	1,100,000			
1955	January	1,370,000			
	February	950,000			
	March	1,460,000			

Table 4. Composition of Tank 5-6 Cell Drainage Waste from 221-T Plant. (3 sheets)

Year	Month	Liters	Pu Grams	Total Beta Activity Curies	Comment
1955	April	1,380,000			
	May	1,410,000			
	June	1,440,000			
The volume and radionuclide content of tank 5-6 cell drainage waste were not recorded separate from other wastes transferred into the cascade of tanks 241-T-110, 241-T-111, and 241-T-112 after July 1954.					

Notes:

(1) HW-11908
(2) HW-20583
(3) HW-25301
(4) HW-33591
(5) Analyses of the combined 2C / 224 building / tank 5-6 waste supernatant stored in tank 241-T-112 conducted on August 6, 1952 and September 24, 1952 indicate that the total beta emitters was comprised of 35 to 50% ruthenium, 35 to 50% cesium, 4 to 8% cerium, yttrium, and other rare earths, and 6 to 11% undetermined (HW-27035, page 8).

4.0 RADIONUCLIDE ANALYSES OF WASTE IN TANK 241-T-104

The U.S. Department of Energy (DOE) uses several factors to determine the disposition of radioactive wastes (DOE M 435.1). One of these factors is the concentration of alpha-emitting transuranic isotopes with half-life greater than 20 years present in the radioactive waste. Two core samples of the waste stored in tank 241-T-104 were obtained in 1992 for chemical and radiochemical analyses. The results of the chemical and radiochemical analyses along with process waste knowledge are were used to determine the inventory of key analytes and radionuclides present in the tank 241-T-104 waste (i.e. best basis inventory).

Table 5 provides the best-basis inventory for the 1C/CW sludge stored in tank 241-T-104, as reported on October 11, 2004 from the Tank Waste Information Network (TWINS) database; http://twins.pnl.gov/twins.htm . The radionuclide inventory is decay corrected to January 1, 2004. The transuranic elements (i.e., Np-237, Pu-238, Pu-239, Pu-240, and Am-241) are summed and presented in Table 6. The concentrations of transuranic elements in the 1C/CW waste stored in tank 241-T-104 are approximately 159.8 ηCi/g. The concentrations of cesium-137 and strontium-90 present in the 1C/CW waste stored in tank 241-T-104 are also provided in Table 6. The cesium-137 and strontium-90 concentrations are approximately 0.155 μCi/g and 2.03 μCi/g.

The inventories of transuranic elements, cesium-137, and strontium-90 present in tank 241-T-104 are also compared to the inventory of these radionuclides present in all 177 underground storage tanks at the Hanford Site in Table 6. The inventory of transuranic elements present in tank 241-T-104 is approximately 0.12 percent of the total inventory of transuranic elements present in all 177 underground storage tanks at the Hanford Site. The inventories of cesium-137 and strontium-90 present in tank 241-T-104 are approximately 0.00056% and 0.0061% of the total inventory of cesium-137 and strontium-90 present in all 177 underground storage tanks at the Hanford Site.

Table 5. Best-Basis Inventory for Tank 241-T-104 Sludge. (3 sheets)

Analyte	Inventory		Basis	Concentration	
106Ru	4.92E-11	Ci	TE	2.87E-14	μCi/g
113mCd	5.78E-02	Ci	TE	3.37E-05	μCi/g
125Sb	7.72E-04	Ci	TE	4.50E-07	μCi/g
126Sn	4.80E-03	Ci	TE	2.80E-06	μCi/g
129I	5.10E-04	Ci	TE	2.98E-07	μCi/g
134Cs	5.89E-07	Ci	TE	3.43E-10	μCi/g
137Cs	2.40E+02	Ci	S	1.55E-01	μCi/g
137mBa	2.27E+02	Ci	C	1.47E-01	μCi/g
14C	< 6.9E-02	Ci	S	4.46E-05	μCi/g
151Sm	1.01E+02	Ci	TE	5.89E-02	μCi/g
152Eu	3.24E-03	Ci	TE	1.89E-06	μCi/g
154Eu	2.11E+00	Ci	S	1.37E-03	μCi/g
155Eu	1.09E+00	Ci	S	7.07E-04	μCi/g
226Ra	6.33E-06	Ci	TE	3.69E-09	μCi/g
227Ac	5.40E-05	Ci	TE	3.15E-08	μCi/g

Table 5. Best-Basis Inventory for Tank 241-T-104 Sludge. (3 sheets)

Analyte	Inventory		Basis	Concentration	
228Ra	7.10E-11	Ci	TE	4.14E-14	µCi/g
229Th	2.00E-08	Ci	TE	1.17E-11	µCi/g
231Pa	4.04E-04	Ci	TE	2.35E-07	µCi/g
232Th	1.68E-10	Ci	TE	9.80E-14	µCi/g
232U	5.66E-06	Ci	C	3.66E-09	µCi/g
233U	4.71E-07	Ci	C	3.04E-10	µCi/g
234U	6.28E-01	Ci	S	4.06E-04	µCi/g
235U	2.03E-02	Ci	S	1.31E-05	µCi/g
236U	6.06E-03	Ci	S	3.92E-06	µCi/g
237Np	2.20E-03	Ci	TE	1.28E-06	µCi/g
238Pu	2.51E+00	Ci	S	1.63E-03	µCi/g
238U	4.64E-01	Ci	S	3.00E-04	µCi/g
239Pu	1.93E+02	Ci	S	1.25E-01	µCi/g
240Pu	2.33E+01	Ci	S	1.51E-02	µCi/g
241Am	2.80E+01	Ci	S	1.81E-02	µCi/g
241Pu	7.84E+01	Ci	S	5.07E-02	µCi/g
242Cm	4.69E-03	Ci	C	3.04E-06	µCi/g
242Pu	1.15E-03	Ci	S	7.44E-07	µCi/g
243Am	2.76E-03	Ci	C	1.78E-06	µCi/g
243Cm	5.24E-05	Ci	C	3.39E-08	µCi/g
244Cm	1.18E-03	Ci	C	7.63E-07	µCi/g
3H	3.75E-01	Ci	TS	2.19E-04	µCi/g
59Ni	1.24E-02	Ci	TE	7.25E-06	µCi/g
60Co	7.91E-02	Ci	TE	4.61E-05	µCi/g
63Ni	1.72E+00	Ci	TE	1.01E-03	µCi/g
79Se	1.27E-03	Ci	TE	7.42E-07	µCi/g
90Sr	3.14E+03	Ci	S	2.03E+00	µCi/g
90Y	3.14E+03	Ci	C	2.03E+00	µCi/g
93mNb	1.47E+00	Ci	TE	8.59E-04	µCi/g
93Zr	1.63E+00	Ci	TE	9.51E-04	µCi/g
99Tc	9.74E-01	Ci	S	6.30E-04	µCi/g
Al	2.51E+04	kg	S	1.62E+04	µg/g
Bi	2.92E+04	kg	S	1.89E+04	µg/g
Ca	2.24E+03	kg	S	1.45E+03	µg/g
Cl	1.04E+03	kg	S	6.70E+02	µg/g
Cr	1.39E+03	kg	S	9.01E+02	µg/g
F	1.33E+04	kg	S	8.57E+03	µg/g
Fe	1.39E+04	kg	S	9.02E+03	µg/g
Hg	1.94E-01	kg	S	1.25E-01	µg/g
K	1.38E+02	kg	S	8.90E+01	µg/g
La	1.69E+00	kg	TS	9.86E-01	µg/g
Mn	9.56E+01	kg	S	6.18E+01	µg/g
Na	9.98E+04	kg	S	6.45E+04	µg/g
Ni	1.75E+01	kg	S	1.13E+01	µg/g
NO2	6.55E+03	kg	S	4.24E+03	µg/g

Table 5. Best-Basis Inventory for Tank 241-T-104 Sludge. (3 sheets)

Analyte	Inventory		Basis	Concentration	
NO3	8.97E+04	kg	S	5.80E+04	µg/g
Oxalate	1.01E+03	kg	C	5.88E+02	µg/g
Pb	7.70E+01	kg	S	4.98E+01	µg/g
PO4	1.14E+05	kg	S	7.35E+04	µg/g
Si	1.01E+04	kg	S	6.52E+03	µg/g
SO4	5.92E+03	kg	S	3.83E+03	µg/g
Sr	1.53E+02	kg	S	9.91E+01	µg/g
TIC as CO3	7.73E+02	kg	S	5.00E+02	µg/g
TOC	5.23E+02	kg	TS	3.05E+02	µg/g
UTOTAL	1.39E+03	kg	S	8.97E+02	µg/g
Zr	1.04E+02	kg	S	6.75E+01	µg/g

Notes:

Radionuclides are decay corrected to January 1, 2004
S – Sample based
C – Calculated
TE – Based on a Hanford Defined Waste model or engineering based waste template
TS – Based on a sample based waste template

Table 6. Transuranic Elements and Fission Products in Tank 241-T-104.

Tank	TRU		Cs-137		Sr-90	
	ηCi/g	Ci	µCi/g	Ci	µCi/g	Ci
241-T-104	159.8	246.8	0.155	240	2.03	3,140
All 177 Tanks	Not applicable	214,067	Not applicable	43,000,000	Not applicable	51,900,000
241-T-104 waste as a percentage of all 177 tanks		0.12%		5.6E-04%		6.1E-03%

Note: TRU = transuranic

5.0 SUMMARY

Tank 241-T-104 received only first decontamination cycle (1C) waste and coating removal waste (CW) from operations of the bismuth phosphate process conducted in the 221-T Plant. The transfer of 1C/CW waste into tank 241-T-104 was conducted periodically from March 11, 1946 through October 19, 1956.

The pH of the 1C/CW waste was adjusted to approximately pH 7 in the 221-T Plant before transfer to the single-shell tanks. This was done to cause the precipitation of bismuth and plutonium in the 1C/CW waste so that the supernatant would contain a lower concentration of plutonium. As a result, tank 241-T-104 contained settled 1C/CW solids (i.e., bismuth and plutonium precipitate) and 1C/CW supernatant.

The 1C/CW sludge was allowed to settle in tank 241-T-104. The 1C/CW supernatant was removed from tank 241-T-104 and either processed in the 242-T Evaporator (April through July 1951) or disposed in the east section of trench 216-T-14 (January 14, 1954). The interstitial liquid was removed from the 1C/CW sludge present in tank 241-T-104 and transferred to other underground storage tanks in two campaigns conducted February 1976 to August 1977 and March 24, 1996 to May 30, 1999.

The concentration of transuranic elements present in the 1C/CW sludge contained in tank 241-T-104 is approximately 159.8 ηCi/g. The concentrations of cesium-137 and strontium-90 in the 1C/CW sludge contained in tank 241-T-104 are approximately 0.155 μCi/g and 2.03 μCi/g, decay corrected to January 1, 2004.

6.0 REFERENCES

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APPENDIX A

VOLUME OF WASTE IN TANK 241-T-104

January 1945 through May 1977

Table A-1. VOLUME OF WASTES IN TANK 241-T-104

Year	Month	Percentage filled	Reference	Page	Comments
1945	January	Empty	HW-7-1293-DEL		No waste transferred into tank 241-T-104.
	February	Empty	HW-7-1388-DEL	18	No waste transferred into tank 241-T-104. 1C/CW waste from 221-T Plant collected into cascade of tanks 241-T-107, 241-T-108, and 241-T-109.
	March	Empty	HW-7-1544-DEL	21	Same as above.
	April	Empty	HW-7-1649-DEL	20	Same as above.
	May	Empty	HW-7-1793-DEL	22	Same as above.
	June	Empty	HW-7-1981-DEL	23	Same as above.
	July	Empty	HW-7-2177-DEL	22	Same as above.
	August	Empty	HW-7-2361-DEL	21	Same as above.
	September	Empty	HW-7-2548-DEL	22	Same as above.
	October	Empty	HW-7-2706-DEL	21	No waste transferred into tank 241-T-104. 1C/CW waste being adjusted to pH 7 before discharge to cascade of tanks 241-T-107, 241-T-108, and 241-T-109.
	November	Empty	HW-7-2957-DEL	21	Same as above.
	December	Empty	HW-7-3171-DEL	21	Same as above.
					Same as above.
1946	January	0% [1]	HW-7-3378-DEL	24	Same as above.
	February	0% [1]	HW-7-3566-DEL	21	Same as above.
	March	3.1% [1]	HW-7-3751-DEL	20 - 21	Tanks 241-T-107, 241-T-108, and 241-T-109 are filled with 1C/CW waste. 1C/CW waste from 221-T Plant diverted to tank 241-U-110 on March 11, 1946. However, transfer line developed a plug. 1C/CW waste was then diverted to tank 241-T-104.
	April	12.9% [1]	HW-7-4004-DEL	20 - 21	1C/CW waste transfer line from 221-T Plant to 241-U Farm (tank 241-U-110) was unplugged. 1C/CW waste from 221-T Plant still being collected in tank 241-T-1104.
	May	19.7% [1]	HW-7-4193-DEL	21	Receiving 1C/CW waste from 221-T Plant into tank 241-T-104
	June	29.2% [1]	HW-7-4343-DEL	23	Receiving 1C/CW waste from 221-T Plant into tank 241-T-104
	July	100% [2]	HW-7-4542-DEL	21 - 22	Receiving 1C/CW waste from 221-T Plant into tank 241-T-104. Tank 241-T-104 filled and 1C/CW waste diverted to tank 241-U-110 on July 22, 1946. Completed on July 22, 1946 the tie-in of new underground pipeline from diversion box 241-T-153 to tank 241-T-105, which permitted the diversion of 2C waste into tank 241-T-105.

Table A-1. VOLUME OF WASTES IN TANK 241-T-104

Year	Month	Percentage filled	Reference	Page	Comments
	August	100% [2]	HW-7-4739-DEL	23	Tank 241-T-104 filled. 1C/CW waste from 221-T Plant diverted to cascade of tanks 241-U-110, 241-U-111, and 241-U-112. Tanks 241-T-105 and 241-T-106 being used to receive 2C waste from 221-T Plant.
	September	100% [2]	HW-7-5194-DEL	26	Same as above.
	October	100% [2]	HW-7-5362-DEL	28	Same as above.
	November	100% [2]	HW-7-5505-DEL	28	Same as above.
	December	100% [2]	HW-7-5630-DEL	25	Same as above.
		100% [2]			Same as above.
1947	January	100% [2]	HW-7-5802-DEL	26	Same as above.
	February	100% [2]	HW-7-5944-DEL	25	Same as above.
	March	100% [2]	HW-7-6048-DEL	23 - 24	Same as above. pH of 1C/CW waste supernatant being received in tank 241-U-110 measured to be between 9 and 10, slightly above target value of pH 7.
	April	100% [2]	HW-7-6184-DEL	26	Same as above. Reduced the amount of caustic solution added t the 1C/CW waste in 221-T Plant to lower the pH to the target value of pH 7, which promotes precipitation of bismuth and plutonium.
	May	100% [2]	HW-7-6391-DEL	23 – 24	Tank 241-T-104 filled. 1C/CW waste from 221-T Plant diverted to cascade of tanks 241-U-110, 241-U-111, and 241-U-112. Tanks 241-T-105 and 241-T-106 being used to receive 2C waste from 221-T Plant.
	June	100% [2]	HW-7-7454-DEL	26	Same as above.
	July	100% [2]	HW-7283-DEL	26	Same as above.
	August	100% [2]	HW-7504-DEL	27	Same as above.
	September	100% [2]	HW-7795-DEL	27	Same as above.
	October	100% [2]	HW-7997-DEL	27	Same as above.
	November	100% [2]	HW-8267-DEL	29	Same as above.
	December	100% [2]	HW-8438-DEL	27	Same as above.

Table A-1. VOLUME OF WASTES IN TANK 241-T-104

Year	Month	Percentage filled	Reference	Page	Comments
1948	January	100% [2]	HW-8931-DEL	28	Tank 241-T-104 filled. 1C/CW waste from 221-T Plant diverted to cascade of tanks 241-U-110, 241-U-111, and 241-U-112. Tanks 241-T-105 and 241-T-106 being used to receive 2C waste from 221-T Plant.
	February	100% [2]	HW-9191-DEL	29 - 30	Same as above.
	March	100% [2]	HW-9595-DEL	32	Tank 241-T-104 filled. 1C/CW waste from 221-T Plant diverted to cascade of tanks 241-U-110, 241-U-111, and 241-U-112. Tanks 241-T-105 and 241-T-106 filled with 2C waste from 221-T Plant.
	April	67.1% [3]	HW-9922-DEL	31 - 32	Tank 241-T-104 filled. 1C/CW waste from 221-T Plant diverted to cascade of tanks 241-U-110, 241-U-111, and 241-U-112. Cribbed 360,000-gallons of 2C supernatant from tank 241-T-105. Tank 241-T-106 still contains 2C waste.
	May	76.2% [3]	HW-10166-DEL	33	241-U-110, 241-U-111, and 241-U-112 are filled with 1C/CW waste. 1C/CW waste from 221-T Plant diverted to tank 241-T-105. Tank 241-T-104 filled with 1C/CW waste. Tank 241-T-106 still contains 2C waste.
	June	85.8% [3]	HW-10378-DEL	30	Same as above.
	July	69.0% [1]	HW-10714-DEL	32 - 33	Same as above. Starting jetting 2C waste from tank 241-T-106 to crib.
	August	70.3% [1]	HW-10993-DEL	35 - 36	Completed jetting 2C waste from tank 241-T-106 to crib on August 3, 1948. Tank 241-T-105 receiving 1C/CW waste from 221-T Plant.
	September	74.0% [1]	HW-11226-DEL	33	1C/CW waste from 221-T Plant diverted to tank 241-T-105, which cascades to tank 241-T-106. Tank 241-T-104 filled with 1C/CW waste.
	October	79.0% [1]	HW-11499-DEL	34	Same as above.
	November	85.0% [1]	HW-11835-DEL	36	Same as above.
	December	94.0% [1]	HW-12086-DEL	37	Same as above.

Table A-1. VOLUME OF WASTES IN TANK 241-T-104

Year	Month	Percentage filled	Reference	Page	Comments
1949	January	100% [1]	HW-12391-DEL	38 - 39	Cascade of tanks 241-T-104, 241-T-105, and 241-T-106 filled with 1C/CW waste from 221-T Plant. Jumper changes made in diversion boxes 241-TX-153, 241-TX-154, and 241-TX-155 to divert 1C/CW waste from 221-T Plant to cascade of tanks 241-TX-109, 241-TX-110, 241-TX-111., and 241-TX-112.
	February	100% [1]	HW-12666-DEL	35	Cascade of tanks 241-T-104, 241-T-105, and 241-T-106 filled with 1C/CW waste from 221-T Plant.
	March	100% [1]	HW-12937-DEL	40 – 41	Same as above.
	April	100% [1]	HW-13190-DEL	40	Same as above.
	May	100% [1]	HW-13561-DEL	42	Same as above.
	June	100% [1]	HW-13793-DEL	41	Same as above.
	July	100% [1]	HW-14043-DEL	43	Same as above.
	August	100% [1]	HW-14338-DEL	44	Same as above.
	September	100% [1]	HW-14596-DEL	43	Same as above.
	October	100% [1]	HW-14916-DEL	43	Same as above.
	November	100% [1]	HW-15267-DEL	45	Same as above.
	December	100% [1]	HW-15550-DEL	43	Same as above.
1950	January	100% [1]	HW-15843-DEL	45	Same as above.
	February	100% [1]	HW-17056-DEL	45	Same as above.
	March	100% [1]	HW-17410-DEL	49	Same as above.
	April	100% [1]	HW-17660-DEL	47	Same as above.
	May	100% [1]	HW-17971-DEL	45	Same as above.
	June	100% [1]	HW-18221-DEL	45	Same as above.
	July	100% [1]	HW-18473-DEL	46	Same as above.
	August	100% [1]	HW-18740-DEL	50	Same as above.
	September	100% [1]	HW-19021-DEL	49	Same as above.
	October	3,170,000-gallons of 1C/CW waste in tanks T-104 thru T-109	HW-19325-DEL	50	Same as above.
	November	3,170,000-gallons of 1C/CW waste in tanks T-104 thru T-109	HW-19622-DEL	49	Same as above.
	December	3,170,000-gallons of 1C/CW waste in tanks T-104 thru T-109	HW-19842-DEL	51	Same as above.

Table A-1. VOLUME OF WASTES IN TANK 241-T-104

Year	Month	Percentage filled	Reference	Page	Comments
1951	January	3,170,000-gallons of 1C/CW waste in tanks T-104 thru T-109	HW-20161-DEL	50	Same as above.
	February	3,170,000-gallons of 1C/CW waste in tanks T-104 thru T-109	HW-20438-DEL	50	Same as above.
	March	3,145,000-gallons of 1C/CW waste in tanks T-104 thru T-109	HW-20671-DEL	54 - 56	Transferred about 25,000-gallons of 1C/CW waste from one of the T-Farm tanks to TX tank in preparation for evaporation in the 242-T Evaporator.
	April	2,055,000-gallons of 1C/CW waste in tanks T-104 thru T-109	HW-20991-DEL	52 - 53	Transferred about 1,115,000-gallons of 1C/CW waste from tanks 241-T-104, 241-T-105, and 241-T-106 to tanks 241-TX-117 and 241-TX-118 in preparation for evaporation in the 242-T Evaporator. An estimated 470,000-gallons of sludge remain in tanks 241-T-104, 241-T-105, and 241-T-106.
	May	1,770,000-gallons of 1C/CW waste in tanks T-104 thru T-109	HW-21260-DEL	56 - 58	242-T Evaporator started up in later part of April 1951. A total of 189,046-gallons of 1C/CW waste processed through May 1948. A total of 1,379,000-gallons of 1C/CW waste transferred from 241-T Farm to 241-TX farm as feed for 242-T evaporator.
	June	1,345,000-gallons of 1C/CW waste in tanks T-104 thru T-109	HW-21506-DEL	55 - 57	A total of 406,568-gallons of 1C/CW waste processed in June 1948 in the 242-T Evaporator. A total of 1,908,625-gallons of 1C/CW waste transferred from 241-T Farm to 241-TX farm as feed for 242-T Evaporator.
	July	792,000-gallons of 1C/CW waste in tanks T-104 thru T-109	HW-21802-DEL	41 - 42	A total of 539,083-gallons of 1C/CW waste processed in July 1948 in the 242-T Evaporator. A total of 2,296,125-gallons of 1C/CW waste transferred from 241-T Farm to 241-TX farm as feed for 242-T Evaporator. This completes the processing of settled 1C/CW waste supernatant from 241-T farm in the 242-T Evaporator.
	August	Not Reported	HW-22075-DEL		
	September	Not Reported	HW-22304-DEL		
	October	Not Reported	HW-22610-DEL		
	November	Not Reported	HW-22875-DEL		
	December	Not Reported	HW-23140-DEL		

[1] Percentage of tanks 241-T-104, 241-T-105, and 241-T-106 filled with waste. Three tanks combined can retain nominally 1,590,000-gallons of waste.
[2] Percentage of tank 241-T-104 filled with waste. Tank can retain nominally 530,000-gallons of waste.
[3] Percentage of tanks 241-T-104 and 241-T-105 filled with waste. Tanks can retain nominally 1,060,000-gallons of waste.

Table A-1. VOLUME OF WASTE IN TANKS 241-T-104

Year	Month	Supernatant (Gallons)	Sludge (Gallons)	Reference	Page	Comments [In general, these are the comments reporting the reference document]
1952	January	Not Reported	Not Reported	HW-23437-DEL		
	February	Not Reported	Not Reported	HW-23698-DEL		
	March	Not Reported	Not Reported	HW-23982-DEL		
	April	530,000	Not Reported	HW-27838	10	
	May	530,000	Not Reported	HW-27838	21	
	June	530,000	Not Reported	HW-27838	32	Cascade of tanks 241-T-104, 241-T-105, and 241-T-106 were filled on March 31, 1952. This is the next and last cascade of 1C/CW waste to be evaporated in the 200 West Area. Evaporation is planned to occur after sufficient allowance for aging (radioactivity decay).
	July	530,000	Not Reported	HW-27839	10	Contents of cascade will be aged one year (to decay shot-lived radionuclides), until March 31, 1953.
	August	530,000	Not Reported	HW-27839	21	Contents of cascade will be aged one year (to decay shot-lived radionuclides), until March 31, 1953, after which time evaporation will be started.
	September	530,000	Not Reported	HW-27839	32	Same as above.
	October	530,000	Not Reported	HW-27840	10	Same as above.
	November	530,000	Not Reported	HW-27840	21	Same as above.
	December	530,000	Not Reported	HW-27840	32	Same as above.
1953	January	530,000	Not Reported	HW-27841	10	Sludge measurements taken.
	February	153,000	377,000	HW-27842	10	
	March	153,000	377,000	HW-27775	10	
	April	153,000	377,000	HW-28043	5	
	May	153,000	377,000	HW-28377	5	
	June	153,000	377,000	HW-28712	5	
	July	153,000	377,000	HW-29054	5	
	August	153,000	377,000	HW-29242	5	
	September	153,000	377,000	HW-29624	5	
	October	153,000	377,000	HW-29905	5	
	November	153,000	377,000	HW-30250	5	

Table A-1. VOLUME OF WASTE IN TANKS 241-T-104

Year	Month	Supernatant (Gallons)	Sludge (Gallons)	Reference	Page	Comments [In general, these are the comments reporting the reference document]
	December	153,000	377,000	HW-30498	5	
1954	January	153,000	377,000	HW-30851	5	Instead of processing in the 242-T Evaporator, Transferred approximately 121,687-gallons of supernatant from tank 241-T-104 to east portion of trench 241-T-1 (renumbered to 216-T-14) on January 15, 1954. Estimated inventories of Cs-137, Sr-90, Pu-239, and U disposed to trench were 230-curies, 1.0-curies, 0.36-grams, and 14,500-grams, respectively (HW-33591, page 12 and HW-38562, page 28). Supernatant in tanks 241-T-105 and 241-T-106 were also discharged to trenches.
	February	66,000	377,000	HW-31126	5	Started receiving T-Plant 1C/CW waste on February 23, 1954.
	March	169,000	377,000	HW-31374	5	Tank 241-T-104 is filled to approximately 6-inches above overflow. Started cascading to tank 241-T-105. Pump in tank 241-T-106 that discharges to trench has a broken shaft and needs to be repaired.
	April	104,000	442,000	HW-31811	5	Tank 241-T-104 is filled to approximately 6-inches above overflow. T-Plant active 1C/CW waste cascade. 1C/CW waste scheduled to be pumped to trench in May 1954, but not conducted.
	May	68,000	478,000	HW-32110	5	Same as above.
	June	38,000	508,000	HW-32389	5	T-Plant active 1C/CW waste cascade. Tank 241-T-106 started to receive 1C/CW waste from cascade on June 17, 1954.
	July	104,000	442,000	HW-32697	5	T-Plant active 1C/CW waste cascade.
	August	5,000	525,000	HW-33002	5	
	September	5,000	525,000	HW-33396	5	
	October	5,000	525,000	HW-33544	5	Plan to pump waste in cascade to tank 241-TX-118. Started scavenging 1C waste in T Plant on October 20, 1954 to precipitate Cs-137 and Sr-90 before discharge of supernatant to single-shell tanks (HW-33585-DEL, page Ed-8).
	November	5,000	525,000	HW-33904	5	Pumping supernatant from 241-T-105 to 241-TX-118 to provide space for storage of coating removal waste from T-Plant
	December	5,000	525,000	HW-34412	5	

Table A-1. VOLUME OF WASTE IN TANKS 241-T-104

Year	Month	Supernatant (Gallons)	Sludge (Gallons)	Reference	Page	Comments [In general, these are the comments reporting the reference document]
1955	January	5,000	525,000	HW-35022	5	
	February	5,000	525,000	HW-35628	5	
	March	5,000	525,000	HW-36001	5	
	April	5,000	525,000	HW-36553	5	
	May	5,000	525,000	HW-37143	5	
	June	5,000	525,000	HW-38000	5	
	July	5,000	525,000	HW-38401	5	
	August	5,000	525,000	HW-38926	5	
	September	5,000	525,000	HW-39216	5	
	October	5,000	525,000	HW-39850	5	
	November	5,000	525,000	HW-40208	5	
	December	5,000	525,000	HW-40816	5	
1956	January	5,000	525,000	HW-41038	5	
	February	5,000	525,000	HW-41812	5	
	March	5,000	525,000	HW-42394	5	
	April	5,000	525,000	HW-42993	5	
	May	5,000	525,000	HW-43490	5	
	June	5,000	525,000	HW-43895	5	
	July	5,000	525,000	HW-44860	5	
	August	5,000	525,000	HW-45140	5	
	September	5,000	525,000	HW-45738	5	
	October	5,000	525,000	HW-46382	5	
	November	5,000	525,000	HW-47052	5	
	December	5,000	525,000	HW-47460	5	
1957	January	15,000	525,000	HW-48144	5	Latest electrode reading of sludge and liquid.
	February	13,000	525,000	HW-48846	5	Latest electrode reading of sludge and liquid.
	March	16,000	525,000	HW-49523	5	Latest electrode reading of sludge and liquid.
	April	18,000	523,000	HW-50127	5	
	May	13,000	525,000	HW-50617	5	Latest electrode reading of sludge and liquid.
	June	13,000	525,000	HW-51348	5	
	July	13,000	525,000	HW-51858	5	
	August	13,000	525,000	HW-52414	5	

Table A-1. VOLUME OF WASTE IN TANKS 241-T-104

Year	Month	Supernatant (Gallons)	Sludge (Gallons)	Reference	Page	Comments [In general, these are the comments reporting the reference document]
	September	13,000	525,000	HW-52932	5	
	October	13,000	525,000	HW-53573	5	
	November	13,000	525,000	HW-54067	5	
	December	13,000	525,000	HW-54519	5	
1958	January	13,000	525,000	HW-54916	5	
	February	13,000	525,000	HW-55264	5	
	March	13,000	525,000	HW-55630	5	
	April	13,000	525,000	HW-55997	5	
	May	13,000	525,000	HW-56357	5	
	June	13,000	525,000	HW-56761	5	
	July	13,000	525,000	HW-57122	5	
	August	13,000	525,000	HW-57550	5	
	September	13,000	525,000	HW-57711	5	
	October	13,000	525,000	HW-58201	5	
	November	13,000	525,000	HW-58579	5	
	December	13,000	525,000	HW-58831	5	
1959	January	13,000	525,000	HW-59204	5	
	February	13,000	525,000	HW-59586	5	
	March	13,000	525,000	HW-60065	5	
	April	13,000	525,000	HW-60419	5	
	May	13,000	525,000	HW-60738	5	
	June	13,000	525,000	HW-61095	5	
	July	13,000	525,000	HW-61582	5	
	August	13,000	525,000	HW-61952	5	
	September	13,000	525,000	HW-62421	5	
	October	13,000	525,000	HW-62723	5	
	November	13,000	525,000	HW-63083	5	
	December	13,000	525,000	HW-63559	5	
1960	January	13,000	525,000	HW-63896	5	
	February	13,000	525,000	HW-64373	5	
	March	13,000	525,000	HW-64810	5	
	April	13,000	525,000	HW-65272	5	

Table A-1. VOLUME OF WASTE IN TANKS 241-T-104

Year	Month	Supernatant (Gallons)	Sludge (Gallons)	Reference	Page	Comments [In general, these are the comments reporting the reference document]
	May	13,000	525,000	HW-65643	5	
	June	13,000	525,000	HW-66187	5	
	July	13,000	525,000	HW-66557	5	
	August	13,000	525,000	HW-66827	5	
	September	13,000	525,000	HW-67696	5	
	October	13,000	525,000	HW-67705	5	
	November	13,000	525,000	HW-68291	5	
	December	13,000	525,000	HW-68292	5	
1961	January thru June	13,000	525,000	HW-71610	5	
	July thru December	13,000	525,000	HW-72625	5	
1962	January thru June	10,000	525,000	HW-74647	5	Latest electrode reading of sludge and liquid.
	July thru December	10,000	525,000	HW-76223	5	
1963	January thru June	10,000	525,000	HW-78279	5	
	July thru December	10,000	525,000	HW-80379	5	
1964	January thru June	10,000	525,000	HW-83308	5	
	July thru December	10,000	525,000	RL-SEP-260	5	
1965	January thru June	44,000	488,000	RL-SEP-659	5	Sludge probably settled in the tank, resulting in reduced volume.
	July thru September	44,000	488,000	RL-SEP-821	5	
	October thru December	44,000	488,000	RL-SEP-923	5	
1966	January thru March	44,000	488,000	ISO-226	5	
	April thru June	44,000	488,000	ISO-404	5	
	July thru September	44,000	488,000	ISO-538	5	
	October thru December	44,000	488,000	ISO-674	5	
1967	January thru March	44,000	488,000	ISO-806	5	
	April thru June	44,000	488,000	ISO-967	5	
	July thru September	44,000	488,000	ARH-95	6	
	October thru December	44,000	488,000	ARH-326	6	
1968	January thru March	44,000	488,000	ARH-534	6	
	April thru June	44,000	488,000	ARH-721	6	
	July thru September	44,000	488,000	ARH-871	6	
	October thru December	44,000	488,000	ARH-1061	7	
1969	January thru March	43,000	488,000	ARH-1200 A	7	

Table A-1. VOLUME OF WASTE IN TANKS 241-T-104

Year	Month	Supernatant (Gallons)	Sludge (Gallons)	Reference	Page	Comments [In general, these are the comments reporting the reference document]
	April thru June	43,000	488,000	ARH-1200 B	7	
	July through September	0	483,000	ARH-1200 C	7	Transferred 48,000-gallons of supernatant to tank 241-TY-103. Waste from other T Farm tanks collected in tank 241-TY-103 was transferred to tank 241-TX-118 and then processed in the 242-T Evaporator.
	October thru December	0	483,000	ARH-1200 D	7	
1970	January thru March	0	483,000	ARH-1666 A	7	
	April thru June	0	483,000	ARH-1666 B	7	
	July thru September	0	482,000	ARH-1666 C	7	
	October thru December	0	483,000	ARH-1666 D	7	
1971	January thru March	0	482,000	ARH-2074 A	7	
	April thru June	0	483,000	ARH-2074 B	7	
	July thru September	0	483,000	ARH-2074 C	7	
	October thru December	0	483,000	ARH-2074 D	7	
1972	January thru March	0	483,000	ARH-2456 A	6	
	April thru June	0	483,000	ARH-2456 B	6	
	July thru September	0	483,000	ARH-2456 C	6	
	October thru December	0	483,000	ARH-2456 D	6	
1973	January thru March	0	483,000	ARH-2794 A	6	
	April thru June	0	483,000	ARH-2974 B	6	
	July thru September	0	483,000	ARH-2974 C	6	
	October thru December	0	483,000	ARH-2974 D	6	
1974	January thru March	0	483,000	ARH-CD-133 A	6	
	April thru June	0	483,000	ARH-CD-133 B	6	
	July thru September	0	483,000	ARH-CD-133 C	6	
	October thru December	0	483,000	ARH-CD-133 D	6	Quarterly report states tank 241-T-104 is "salt filled".
1975	January thru March	0	483,000	ARH-CD-336 A	6	Quarterly report states tank 241-T-104 is "salt filled".
	April thru June	0	483,000	ARH-CD-336 B	6	
	July thru September	0	483,000	ARH-CD-336 C	6	Quarterly report states tank 241-T-104 is "salt filled".
	October thru December	0	483,000	ARH-CD-336 D	6	Quarterly report states tank 241-T-104 is "salt filled".
1976	January thru March	0	483,000	ARH-CD-702 A	6	"Salt filled." Transferred 15,000-gallons of saltwell liquor to tank 241-T-101.

Table A-1. VOLUME OF WASTE IN TANKS 241-T-104

Year	Month	Supernatant (Gallons)	Sludge (Gallons)	Reference	Page	Comments [In general, these are the comments reporting the reference document]
	April thru June	0	483,000	ARH-CD-702 B	6	Removed from service. Salt filled.
	September	0	483,000	ARH-CD-702 I	14	Tank is inactive and salt filled. Saltwell pumping conducted.
	October	0	483,000	ARH-CD-822-OCT	15	Tank is inactive and salt filled. Saltwell pumping conducted.
	November	0	483,000	ARH-CD-822-NOV	15	Tank is inactive. Saltwell pumping conducted.
	December	0	483,000	ARH-CD-822-DEC	17	Tank is inactive. Saltwell pumping conducted.
1977	January	0	483,000	ARH-CD-822-JAN	17	Tank is inactive. Saltwell pumping conducted.
	February	0	483,000	ARH-CD-822-FEB	17	Tank is inactive. Saltwell pumping conducted.
	March	0	483,000	ARH-CD-822-MAR	17	Tank is inactive. Saltwell pumping conducted.
	April	0	483,000	ARH-CD-822-APR	17	Tank is inactive. Saltwell pumping conducted.
	May	0	483,000	ARH-CD-822-MAY	17	Tank is inactive. Saltwell pumping conducted.

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CH-TRUM WPU&SE 11-Tank Material Balance

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Abstract: This document releases the architect-engineer material balance for the Transuranic Waste Project Packaging facility, specifically for processing the eleven contact-handled waste tanks.

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EXECUTIVE SUMMARY

The waste from the B-200 and T-200 series tanks and the waste contained in T-104, T-110 and T-111 will be retrieved by the Single-Shelled Tank Waste Retrieval System and will be transferred to the Contact-Handled Transuranic/Mixed Waste Packaging Unit & Support Equipment. Two large dryer systems will process the approximately 1.4 million gallons of waste contained in these tanks. This Contact-Handled Transuranic/Mixed Waste Packaging Unit & Support Equipment flowsheet was developed to determine the properties of the dried product and the liquid effluent by-product and to compare these properties with waste acceptance criteria for the Waste Isolation Pilot Plant and Liquid Waste Processing Facilities. These properties include Transuranic Concentration, Fissile Gram Equivalent Activity, Plutonium Equivalent Activity, Dose Equivalent Rate, and Decay Heat in the dried product, and concentrations of certain constituents in the liquid effluent. Strontium-90 and Cesium-137 content in the final product was also determined for dosage analysis.

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J. Excerpt from *Integrated Emissions Baseline Report,* RPT-W375-ES000001, Rev 1

K. Excerpt from *Hanford Site Solid Waste Acceptance Criteria*, HNF-EP-0063 Rev 10, Table A-1 and Table B-1

L. Email from Ivan Papp to John Garfield, *242-A Evaporator Partitioning*, with excerpt from attached file 242-A Evaporator Partitioning .xls

M. Excerpt from *Tank Farm Contractor Operation and Utilization Plan*, HNF-SD-WM-SP-012

N. Excerpt from *Contact-Handled Transuranic Waste Acceptance Criteria for the Waste Isolation Pilot Plant*, DOE/WIPP-02-3122, Rev 0.1 Table B-1

O. Email from Mark Bowman to Paul Day, *Waste Acceptance Document ST4500 Constituents and LERF Rad Limits* with attached file, List of Constituents.xls

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LIST OF TERMS

BBI	Best Basis Inventory
CH-TRUM	Contact Handled Transuranic Mixed
DE-Ci	Dose Equivalent Curies
ETF	Effluent Treatment Facility
FGE	Fissile Gram Equivalent
FRPS	Feed Receipt Process System
PE-Ci	Plutonium-239 Equivalent Curies
SST	Single-Shell Tank
TOE	Total Operating Efficiency
TRU	Transuranic
TWINS	Tank Waste Information Network System
wt%	weight percent
WIPP	Waste Isolation Pilot Plant
WPU&SE	Waste Packaging Unit & Support Equipment
WRS	Waste Retrieval System

1.0 OBJECTIVE/PURPOSE

1.1 PURPOSE

The purpose of this calculation is to perform a material balance around each of the contact-handled transuranic/mixed (CH-TRUM) Waste Packaging Unit and Support Equipment (WPU&SE) components. This is done on an 11-tank average basis, an individual tank basis, and a dry bed basis. The 11-tank average material balance provides an overall view of the CH-TRUM WPU&SE process, based on operating parameters for each unit component, to determine if the process meets end-product criteria. The individual tank basis analysis defines minimum and maximum values for various key process parameters and key analytes. The dry bed analysis investigates the effect of drying and the addition of additives on end-product criteria by performing a 55-gallon drum (drum)-by-drum material balance around the dryer.

1.2 SCOPE

Output from the Single-Shell Tank (SST) Waste Retrieval System (WRS) is the input for the CH-TRUM WPU&SE. To assist the retrieval process, the SST WRS may retrieve waste from the tanks with a range of water dilution values. Since the CH-TRUM WPU&SE process uses dryers to separate water from the waste prior to packaging, the water content used for retrieval will impact the estimated processing time and system throughput. The material balance assumes a 1:1 volume ratio of water to in-tank waste (undiluted bulk sludge phase) [Assumption #1]. Nominal WPU&SE process throughput is 6,000 gallons of undiluted tank waste per day with two dryers at a 100% Total Operating Efficiency (TOE) [Design Input #20]. At a 1:1 dilution and a 67% TOE, this is 8,040 gallons of slurry processed per day.

An average of 50% of the dilution water used by the SST WRS will be recycled liquid effluent produced by the CH-TRUM WPU&SE process (Assumption #20). Recycling of the liquid effluent has the potential for build up of volatile constituents over time. This calculation estimates the volume of liquid effluent returning to the SST WRS, but does not account for the retreatment of recycled components. A scoping calculation was performed (Reference #4) that verifies that the recycling buildup of certain constituents, with the exception of Tritium for Tank B-201, will not hinder the ability of the liquid effluent to meet Effluent Treatment Facility (ETF) waste acceptance criteria. The Tritium value for B-201 is 10% higher than the ETF limit and could be resolved by limiting the percent recycle or verifying the Best Basis Inventory (BBI) value for Tritium.

2.0 SUMMARY OF RESULTS AND CONCLUSIONS

Transuranic (TRU) concentration, Pu-239 Fissile Gram Equivalent (FGE) activity, Pu-239 Equivalent (PE-Ci) activity, Decay Heat, and Dose Equivalent (DE-Ci) activity of the dried product were analyzed on an 11-tank average basis, an individual tank basis, and a dry bed drum-by-drum basis. These values were then compared to limits called out in the waste acceptance criteria for the dried product.

The dried product meets the WIPP waste acceptance criterion for each drum by containing more than 100 nCi/g of alpha-emitting TRU isotopes (Criterion #8), with an overall average of 586 nCi/g of packaged waste, a tank range average of 249 to 1,735 nCi/g and a drum range of 246 to 1,905 nCi/g.

The Pu-239 FGE was calculated for a comparison to its drum limit. For a drum the Pu-239 FGE plus 2 times the standard deviation must be less than the 200 gram limit (Criterion #7). The dried product meets this criterion by averaging 7.3 Pu-239 FGE per drum on an overall average basis, a range of 1.1 to 16.9 Pu-239 FGE on an individual tank basis, and a range of 1.05 to 16.7 for the drum-by-drum approach.

Like the TRU content and Pu-239 FGE, the drums successfully meet the PE-Ci limit (Criterion #9), Decay Heat limit (Criterion #10), and DE-Ci limit (Criterion #11).

The liquid effluent returned to the SST WRS or sent to ETF for treatment and disposition must meet ETF waste acceptance criteria (Criterion #5). ETF must maintain an Ecology-approved influent constituents list defining all constituents and maximum concentrations acceptable for treatment in the ETF (Reference #18). A comparison of the concentration of analytes in the liquid effluent to this list shows that most analytes are well below these set limits. However, Uranium-234 is the only analyte estimated to be slightly over the limit set by this constituents list.

3.0 INTRODUCTION/BACKGROUND

The CH-TRUM WPU&SE will be used to handle, dry and package waste retrieved from eleven SSTs. The eleven SSTs include the B-200 series, T-200 series, T-104, T-110 and T-111. Waste will be retrieved by a SST WRS and will then be delivered in batches to the CH-TRUM WPU&SE.

The CH-TRUM WPU&SE will utilize two dryer systems, and is sized to process approximately 1.4 million gallons of undiluted waste (nominal 2.8 million gallons diluted). A nominal 6,000 gallons of undiluted tank waste will be processed each day at 100% total operating efficiency (TOE). At 1:1 dilution and a 67% TOE, 8,040 gallons of slurry will be processed through the dryers per day. The slurry will be dried to a target 10 weight percent (wt%) water and will be packaged in drums for final disposal at the Waste Isolation Pilot Plant (WIPP) near Carlsbad, New Mexico. Additive(s) will be used, as necessary, to control buildup of waste on the inside of the dryer, to build a dry bed for dryer startup, and to insure a flowable material that can be transported to the drum. The resulting condensate from the dryer operation will be sent to the ETF by tanker truck or returned to the WRS.

4.0 INPUT DATA

Specific information used for this calculation can be categorized as Design Inputs, Criteria, and Assumptions. Design inputs are typically dictated by the features of equipment, by other design media, or by data in specification documents. Criteria are typically requirements dictated by a guiding specification document or a project-specified goal. Assumptions are inputs needed in order to perform a calculation. The assumptions used in this calculation are provided in Section 5.0.

4.1 DESIGN INPUTS

1. Process Flow Diagram (Appendix A) that identifies process streams and relationships around unit operations.

2. Tank Compositions (Appendix G)

   - BBI values for B-200 Series, T-200 Series, T-110 and T-111, downloaded 2/16/04 (Reference #15);

   - BBI values for T-104, downloaded 6/30/04 (Reference #15); and

   - RPP-17790 (Reference #8), which captures the input to the Hanford Tank Waste Operations Simulator.

3. Waste Wash Factors (Appendix H)

   - BBI values for B-200 Series tanks, T-200 Series tanks, T-110 and T-111, downloaded 2/16/04 (Reference #15);

   - BBI values for T-104, downloaded 6/30/04 (Reference #15); and

   - RPP-17790 (Reference #8), which captures the input to the Hanford Tank Waste Operations Simulator.

4. Undiluted tank volumes, densities and percent water content by phase (Appendix I).

   - BBI values for B-200 Series tanks, T-200 Series tanks, T-110 and T-111, downloaded 2/16/04 (Reference #15); and

   - BBI values for T-104, downloaded 6/30/04 (Reference #15).

5. Release Factors for Waste Treatment Plant (Reference #2, Appendix J).

6. Half-Life, Specific Activity, Decay Heat, DE-Ci Correction Factors, and Pu-239 FGE Conversion Factors (Reference #14, Appendix K).

7. Evaporator and Condenser Partitioning Factors – 242-A Evaporator Data (Appendix L). Two columns were inserted that were calculated from the given data.

8. PE-Ci Weighing Factors (Reference #13, Appendix N).

9. The target dryer fill is 70 vol% to start and then is decreased to 60% (Reference #3).

10. Target product endpoint is 10 wt% Scouring Agent and 10 wt% water (Reference #3).

11. Density Correlation Coefficients (Reference #9, Appendix M).

12. Gross internal volume of a drum is 7.3 ft^3 (Reference #17).

13. Relative Humidity and temperature values for off-gas streams (Reference # 6).

14. Off-gas flow rate for the FRPS tanks is 10 scfm for each of the five tanks. The Dewatering System (DWS) Condensers off-gas flow rate is 150 scfm for each condenser, the WPS ISO-Container off-gas flow rate is 325 scfm (includes conveyor) and the Dryer ISO-Container off-gas flow rate is 650 scfm per compartment.(Reference #6).

15. Average Hanford Site temperature is 45°F based on range of -25°F to 115°F (Reference #16).

16. Average Hanford Site Relative Humidity is 55% (Reference #16).

17. Air volumes at various temperatures (Reference #10).

18. Maximum condensate that will be formed in a drum is 10.7 grams (Reference #5).

19. Each dryer has a net volume of 2,291 gallons (8,672 L) [from dimensions in Reference #12].

20. Nominal process throughput is 6,000 gallons of undiluted tank waste per day with two dryers at a 100% TOE (Reference #7). Therefore, at a 1:1 dilution and 67% TOE (Reference #7), 8,040 gallons per day of slurry will be received at the Feed Receipt Process System (FRPS) tanks.

21. Properties of the Flowability Agent (Vermiculite) and the Scouring Agent (Sand) (Reference #3).

22. 4,910 kg of Flowability Agent is needed for a dry bed startup of two dryers (Reference #3).

23. Waterholding capacity of Vermiculite: 220-325% by weight, or 0.455 kg Vermiculite per kg water (Reference #19).

4.2 CRITERIA

1. The CH-TRUM WPU&SE shall package TRU waste in 55-gallon drums (Reference #7).

2. The CH-TRUM WPU&SE shall be designed to disposition a minimum of 1.4 Mgal of CH-TRUM tank waste retrieved from SSTs. The dewatering system shall be sized to process a nominal 6,000 gallons of undiluted tank waste per day (24 hours) at a total operating efficiency (TOE) of 100% (Reference #7).

3. The CH-TRUM WPU&SE shall be capable of packaging the dewatered waste at a rate equal or greater than the processing rate of the dewatering system (Reference #7).

4. The CH-TRUM WPU&SE shall be designed to have a reliability/availability consistent with a TOE of 67% (Reference #7). The TOE shall include a 30-day period for disassembly after operations, relocation to T-Farm, reassembly and check out (i.e., operational testing performed) and a 30-day period when no waste retrieval or packaging is taking place.

5. Liquid streams generated for return to the SST WRS and to a tanker truck must meet acceptance criteria for the ETF [Reference #7].

6. Maximum gross weight for a filled drum – 1,000 lbs (Reference #13).

7. Fissile Gram Equivalent (FGE) Pu-239 plus 2 times the standard deviation must be less than the 200 gram limit for a 55-gallon drum (Reference #13).

8. Waste payload containers shall contain more than 100 nCi/g of alpha-emitting TRU isotopes with half-lives greater than 20 years (Reference #13).

9. PE-Ci limit for a drum in good condition and directly loaded is less than or equal to 80 (Reference #13).

10. Decay heat limit for storage at the Central Waste Complex is 0.1 watt/ft^3 or 0.73 watts per drum (Reference #14). For acceptance at WIPP, the limit value is based on the assigned shipping category limit specified in Table 5.2-1 of the CH-TRAMPAC. This limit will be determined after final establishment of a shipping category (Reference #13).

11. Dose Equivalent Curie (DE-Ci) limit for a drum is 82.5 (Reference #14).

12. For waste that could form condensate during storage, sufficient sorbent shall be added to the container to sorb any condensate formed (Reference #14).

5.0 ASSUMPTIONS

1. Dilution ratio for the undiluted waste is 1:1 on a volume basis (one volume dilution water added to one volume of undiluted sludge waste from the tank).

2. Tanks will be retrieved in the following order: B-201, B-202, B-203, B-204, T-201, T-202, T-203, T-204, T-110, T-111, T-104.

3. This dried waste has a bulk density of 1.60 kg/L. This density estimate is a conservative value that is calculated using the absolute densities and fractions of the scouring agent (ρ = 2.3, 10 wt%), water (ρ = 1.0, 10 wt%), and the waste solids (ρ = 3.5, 80 wt%), and using an assumed void fraction of 0.40.

$$\frac{1}{\rho} = \sum_{i=1}^{n} \frac{x_i}{\rho_i}$$

4. The weight of the waste packaged in a 55-gallon drum is 620 lb. This is based on the internal volume of a 55-gallon drum, bulk density of the waste of 1.60 kg/L, and a maximum fill volume limit of 85%.

5. For nominal throughput, processes are assumed to operate continuously 24 hours per day. A year is defined as 365 days or the number of processing days, whichever is less. This is done because the amount of waste processed per year is the daily rate multiplied by the number of days in a year. If processing is done in less than 365 days, using a full 365 days is meaningless because it would indicate that more waste was processed than actually projected to be removed from the tanks.

6. Dryer retention factors (i.e., the fraction of a species that is retained in the dryer during processing) are the same as for the 242-A Evaporator Vapor-Liquid Separator. Condenser retention factors are the same as the 242-A Evaporator Primary Condenser Unit. For those species that are not addressed in the 242-A Evaporator data, a partitioning factor of 8.0E+06 is applied to the feed for that constituent. A partitioning factor is the initial mass in the dryer divided by the mass that exits the dryer in the vapor. This value is based on an assumption derived from dryer operating experience.

7. Release factors are used to predict the amount of species entrained in the off-gas of vessels other than the dryer. The release factor for the FRPS tanks is 6.97×10^{-9} (mass/m^3 in air)/ (mass/m^3 in liquid) and the release factor for the Waste Packaging System (WPS) is 5.98×10^{-9} (mass/m^3 in air)/ (mass/m^3 in liquid). This is based on the Waste Treatment Plant baseline release factors for the Low-Activity Waste (LAW) Feed Receipt Vessel and the High-Level Waste (HLW) Feed Blending Vessel (Reference #2). Species entrained in the off-gas for the DWS ISO-Containers is assumed to be negligible.

8. The water content of the flowability agent and scouring agent is negligible.

9. The packaging process does not change the product density.

10. Vermiculite is added to the 55-gallon drum based on the maximum amount of condensate that may be formed inside the 55-gallon drum once the lid is in place. The recommended ratio of vermiculite to water is 0.455 kg of sorbent per kg of water. As a conservative approach, this calculation uses 2 times this amount of vermiculite. Based on the expected amount of condensate (Design Input #18), the vermiculite addition is negligible compared to the overall mass of the dried product in a 55-gallon drum.

11. Any water that accompanies the inlet air passes through the ventilation system and out the stack. This allows for a clearer illustration of how much waste is conveyed to the OGTS.

12. The WPS will produce a 55-gallon drum of Secondary Solid Waste per day.

13. The LES Storage Tanks will be flushed twice during the life of the operation; once before transfer between farms and once at the end of operation. Each tank is estimated to be flushed with 5,000 gallons of filtered water.

14. The Waste Receipt Tanks will be flushed twice during the life of the operation; once before transfer between farms and once at the end of operation. Each tank will be flushed with 1,800 gallons of filtered water. This estimated value is 25% of the working volume of a tank.

15. As a conservative approach, 100% of the off-gas coming into the Off-Gas Treatment System (OGTS) goes out through the stack (i.e., no solid waste is caught in the HEPA filter).

16. The liquid leaving the Liquid Effluent Tanks has no solids over 5 microns that would be caught in the 5-micron filter.

17. The 55-gallon drums will not be overpacked within a standard waste box during normal operating conditions.

18. Based on the number of tanks in the Feed Receipt Process System (five total), any mixing between waste from different SSTs (for example B-201 and B-202) is negligible.

19. Solids loss to the condensate stream of the dry bed method is negligible.

20. 50% of the dilution water needed for retrieval will be recycled back to the WRS.

21. Criteria for the 55-gallon drum headspace gas concentration (i.e., hydrogen/methane, VOC, and total gas), from Section 3.5.5 "Headspace Gas Concentrations" of the *CH-TRU Waste Acceptance Criteria for the Waste Isolation Pilot Plant*, will be met.

6.0 METHODS OF ANALYSIS

The CH-TRUM mass balance is a Microsoft Excel® spreadsheet designed to estimate stream compositions and flow rates based on parameters that can be updated as assumptions are changed or alternatives are evaluated. The mass balance will produce results one tank at a time or for groups of tanks. For the evaluation of the overall performance of the system, all eleven tanks to be retrieved from B-Farm (B-201, B-202, B-203, and B-204) and T-Farm (T-201, T-202, T-203, T-204, T-104, T-110, and T-111) are included in the material balance.

The spreadsheet is set-up with a number of worksheets used to support the material balance. Values for the streams shown on the Process Flow Diagram (Appendix A) are calculated on the worksheet titled "Process Baseline" (Appendix B). The individual tank stream values and tank-specific processing parameters are calculated in worksheets titled "B-201 Process", "B-202 Process" etc, which are in Appendices B-1 to B-11. The effect of utilizing a dry bed approach is shown in Appendix C. Inputs used for the calculation are located on the "Inputs" worksheet (Appendix D). The "SST Data" worksheet (Appendix E) shows the waste composition as it sits in the tank and after retrieval. This composition data is pulled from the worksheets titled "In Tank Inventory" (Appendix G) and "Wash Factors" (Appendix H). The SST Data worksheet is also used to simulate the decay of the radionuclides and the dilution of the waste in each of the tanks. The amount of water added for dilution is a function of the "dilution factor" specified on the Process Baseline worksheet. The data from the SST Data worksheet is "called" from the Process Baseline worksheet using either a "1" multiplier to include the tank or a "0" to exclude the tank from the stream balance, thus enabling the individual tank worksheets (Appendices B-1 to B-11) to be produced.

The "ETF WAC" worksheet summarizes the results from the comparison of the Liquid Effluent stream (Stream 28) to the ETF Waste Acceptance Criteria (Criterion #5). This worksheet is provided in Appendix F. Values for the 242-A Evaporator, used to estimate liquid/solid separation in the dryer unit and liquid/vapor separation in the condenser, are from the source data provided in Appendix L.

The "B-Farm Dry Bed Process" and "T-Farm Dry Bed Process" worksheets are used to simulate the dry bed operating process. Besides demonstrating the dry bed process, the worksheets also summarize the results of the dry bed operation on the end-product criteria. These results include drum composition, TRU concentration, FGE Pu-239, DE-Ci, PE-Ci, and Decay Heat.

7.0 CALCULATION AND ANALYSIS

7.1 DEFINITIONS

The following list contains the definitions for key variables used in the calculation and does not include all variables.

- **Dilution Water** – Water added to the tank waste during retrieval and subsequent transfer to the CH-TRUM WPU&SE. The volume added is based on the ratio of water to **Sludge** on a volume basis.

- **Dry Solid** – Is comprised of the total waste minus all water. The dry solid mass can be taken as a constant for tracking the waste through the system.

- **Free Liquid** – Liquid phase that is comprised of any **Supernate** in the tank plus **Dilution Water** added for retrieval. This is all the diluted liquid that is not **Sludge Liquid**. Composition of the **Free Liquid** is the same as the **Sludge Liquid**.

- **Packaged Product** – Treated waste stream from the dryer unit. This material is comprised of **Dry Solid** plus residual water plus additive material (Vermiculite or Scouring Agent). The residual water percentage is a dryer operating target.

- **Sludge (Bulk Sludge)** – Tank waste phase comprised of both **Sludge Solid** and **Sludge Liquid** material. The volume of this phase is used to calculate the volume of **Dilution Water** added to the waste stream sent to the CH-TRUM WPU&SE system.

- **Sludge Liquid** – Liquid in chemical equilibrium with the **Sludge Solid** material. Liquid occupies the interstitial space in the **Sludge** phase and is not free flowing.

- **Sludge Solid** – Solid material in chemical equilibrium with the **Sludge Liquid** material. Solid material contains water soluble species that can become liberated upon further addition of water to the waste. The degree to which a given species will dissolve into the added water is dictated by a wash factor developed by the Hanford Site contractor.

- **Supernate** – Free flowing liquid phase present in the tank prior to retrieval. This waste phase contains dissolved solid material that will contribute to the **Dry Solid** mass. The volume of this phase is not used to calculate the volume of **Dilution Water**.

- **Total Liquid** – Sum of the **Sludge Liquid** and the **Free Liquid**.

7.2 TANK INVENTORY BASIS

The "In Tank Inventory" worksheet (Appendix G) was constructed to determine the compositional breakdown of the Supernatant and Sludge phases of the waste as it sits in the tank. A "SST Data" worksheet (Appendix E) was developed to estimate the feed to the CH-TRUM WPU&SE based on the In Tank Inventory data, radionuclide decay factors, and retrieval dilution. This worksheet treats each of the tanks individually so that their impact on the material balance can be evaluated separately or in different combinations.

The basis inventories for the mass balance were obtained from the BBI data, located on the Tank Waste Information Network System (TWINS) web page (Reference #15), on February 16, 2004 and June 30, 2004. These values are shown on the "In Tank Inventory" worksheet. The BBI data does not identify certain key analytes, such as hydroxide. Due to the lack of BBI data for these constituents [shown as na on the worksheet], in-tank inventory estimates for those constituents that support the HTWOS model were used. The HTWOS model is the official tank farm retrieval mass balance. These in-tank inventory estimates are documented in *Double and Single Shell Tank Inventory Input to the HTWOS Model – 2003A Update* (Reference #8). The HTWOS in-tank inventories are based on the BBI and reflect tank contents effective as of July 1, 2002, with additions for hydroxide and other sensitive analytes, such as ammonia, that are important to permitting. The BBI data, which reports percent water, density and volume of each phase (supernatant and sludge) in the eleven tanks, was used to calculate the water content. The inventories for isotopes in the document are provided in kilograms, and therefore are converted to Curies using specific activity data (see "Inputs" worksheet).

7.3 RADIONUCLIDE DECAY

Radionuclides decay over time into other constituents. This occurs when an unstable isotope transforms into a different isotope by emitting a subatomic particle, such as an alpha or beta particle or gamma ray. The BBI provides inventory activities for the radionuclides in the eleven tanks with a decay date of January 1, 2001. Therefore, the "SST Data" worksheet (Appendix E) estimates the activity of the radionuclides at the beginning of operations, currently scheduled for March 1, 2005.

The following equation was used:

$$N = N_0 \cdot e^{\frac{-0.693 \cdot t}{T_{1/2}}}$$

where:

N_0 = initial activity

t = time elapsed since the initial activity was measured (days)

$T_{1/2}$ = half-life of isotope (days)

This calculation takes into account the loss of radionuclides over time, but does not consider the daughter products of the waste decay. The activities of Y-90 and Ba-137m, however, were estimated by secular equilibrium. This allows a prediction of the amount of the isotope by

measuring the concentration of another. Y-90 is dependent on the amount of Sr-90, while Ba-137m is dependent on the amount of Cs-137 (Reference #1).

7.4 WASHED FEED DISTRIBUTION

To make the feed data flexible enough to use for drying and to evaluate the impact of a leak, the data for the feed must reflect which species are dissolved in the retrieved waste liquid, which are solid particles, and how much water is present. The SST Data worksheet assembles columns for each tank to calculate the distribution of waste between the total liquid and dry solid phases after dilution.

The Request For Proposal (RFP) indicates that some of the waste in the tanks will dissolve upon addition of water, but does not quantify how much will dissolve. The document, *Double and Single Shell Tank Inventory Input to the HTWOS Model – 2003A Update* (Reference #8), reports wash factors for analytes in the eleven tanks. These factors were used in the SST Data worksheet to determine the mass of waste that dissolves into the liquid phase during the 1:1 dilution (the dilution factor is defined as the volume of water added per volume of wet, in-tank sludge). For example, the tank B-201 silver (Ag^+) analyte mass in the Washed Liquid phase after dilution is calculated by taking the total Decayed Inventory Sludge analyte (SST Data, cell F21) and multiplying it by the Ag^+ wash factor from Appendix H – Wash Factors, and then adding the mass of the analyte from the Decayed Inventory Supernatant (SST Data, cell E21).

The amount of water in the Washed Liquid is determined from the quantity of water present in the waste prior to retrieval plus the amount of water equal in volume to the volume of wet Sludge before dilution. For example, for Tank B-201, the Washed Liquid water is equal to the water in the Decayed Inventory Supernatant (SST Data, cell E20) plus the water in the Decayed Inventory Sludge (SST Data, cell F20) plus the water volume found by multiplying the Decayed Inventory Sludge volume (SST Data, cell F4) by the dilution factor (SST Data, cell G13).

To calculate the total liquid volume, the density of the liquid is needed. This is done by using the following Density Correlation equation from TFCO&UP (Reference #9):

$$d_T = \frac{m_T \cdot d_w}{\left(m_T - 1000 \cdot d_w \sum \left(\frac{\gamma_i \cdot m_i}{mw_i} \right) \right)}$$

Where m_T is the total mass (kg) of the liquid phase, d_w is the density (kg/L) of water, mw_i is the molecular weight (mol/L) of the analyte, and γ_i is the density coefficient of the analyte from Table A-41 of TFCO&UP (shown on the "Inputs" worksheet).

7.5 FEED TO WASTE RECEIPT TANKS

From the in-tank inventory, the SST Data worksheet develops estimates for six categories for each tank. For any assumed dilution factor, the following are computed:

- Total Liquid (sum of Free Liquid and Sludge Liquid);
- Sludge Solid (this is the sludge remaining after washing);
- Sludge Liquid (mass is equal to the difference between Bulk Sludge and Sludge Solid);
- Bulk Sludge (sum of sludge liquid and sludge solid);
- Free Liquid (Total liquid minus Sludge Liquid); and
- Total Waste (Bulk Sludge plus Free Liquid).

Total Liquid is equal to the Washed Liquid, while the Sludge Solid is equal to the Washed Dry Solid. The volume of Bulk Sludge is determined by multiplying the volume of initial sludge times the mass of washed solids to initial dry solids. For Tank B-201, the volume of Bulk Sludge is the mass of Sludge Solid (SST Data, cell J8) divided by the mass of Tank Inventory Dry Sludge (SST Data, cell D8) multiplied by the volume of Decayed Inventory Sludge (SST Data, cell F4). This volume is then used to calculate the sludge density. This density and the sludge volume is used to calculated the mass of the Bulk Sludge.

The mass of Sludge Liquid, by definition, is the difference of Bulk Sludge minus Sludge Solid. The concentration of analytes in the Sludge Liquid will be the same as for the Total Liquid. Therefore, the mass of each analyte in the Sludge Liquid is proportional to the mass fraction of Sludge Liquid to Total Liquid. For example, the mass of Al^{3+} in the Sludge Liquid is equal to the mass of Al^{3+} in the Total Liquid (SST Data, cell I21) multiplied by the ratio of the mass of Sludge Liquid (SST Data, cell K9) divided by the mass of Total Liquid (SST Data, cell I9). Once all of the analytes are calculated for the Sludge Liquid, the mass in the Bulk Sludge is Sludge Solid plus Sludge Liquid. The mass and composition for Free Liquid is therefore the difference between Total Liquid and Sludge Liquid. Total Waste is the sum of Bulk Sludge and Free Liquid.

7.6 STREAM DESCRIPTIONS AND MATERIAL BALANCE RELATIONSHIPS

Stream 1 – FRPS Tanks Air Inlet

Each of the Feed Receipt Process System (FRPS) tanks is equipped with a HEPA-filtered air inlet for venting, which prevents the potential accumulation of flammable and/or hazardous gases in the headspace above the waste. The inlet also provides a filtered exit path in the event of pressurization of the tank from an upset condition. The volumetric flow rate of air through each tank is expected to be 10 scfm (Design Input #14). The calculation assumed that the air entering the FRPS tanks is 45°F (Design Input #15) and has a relative humidity of 55% (Design Input

#15) , and that any water that accompanies the inlet air passes through the ventilation system and out the stack (Assumption #11).

Stream 2 – WRS Slurry Feed to FRPS Tanks

The CH-TRUM WPU&SE will receive slurry feed from the SST WRS via one of two hose-in-hose transfer lines provided by others. The two transfer lines are connected to a header in the FRPS that allows waste to be collected in one of five tanks (waste feed lines to each tank correspond to Stream 2). Waste will be delivered in batches. See Sections 7.2 through 7.5 for explanation of waste feed inventory.

Stream 3 – FRPS Tanks Off-Gas

Each of the FRPS tanks will have an air sweep through the headspace to maintain the tanks at a slight negative pressure relative to atmospheric conditions. This will prevent the potential build-up of hydrogen and noxious gases in the tank headspace and draw any such gases through the monitored OGTS. Material may become entrained in the FRPS Tanks Off-Gas stream as slurry is added to the tanks as a result of mixing of the tank contents or from concentration differences (such as water humidifying the air stream). A release factor of 6.97×10^{-10} (mass/m^3 in air)/ (mass/m^3 in liquid) from the Waste Treatment Plant Low Activity Waste feed receipt tanks (Reference #2) has been used to estimate the quantity of waste entrained in the Waste Receipt Tank Off-Gas (Assumption #7). The total mass of waste lost to the off-gas is calculated by multiplying the release factor, the air flow rate, the density of the waste, and the total number of processing days. Loss of material to this off-gas stream is minimized by the design of waste inlet nozzles and mixing system. Target pressure in the tank headspace is slightly negative with respect to atmosphere with an off-gas flow rate for each tank of 10 scfm (Design Inputs #14). The water content of this stream is determined by the Relative Humidity, which was assumed to be 100%, and a stream temperature of 45°F (Design Inputs #15).

Stream 4 – Slurry Feed to DWS Dryers

The Slurry Feed to the Dewatering System (DWS) Dryers is a batch process with an assumed maximum dryer capacity of 1,604 gallons per dryer. This is based on the 2,291 gallons total volume of each dryer (Design Input #19) and an assumption that the dryer will be operated at a maximum of 70% filled (Design Input #9). The composition of the dryer feed is based on the slurry feed stream from the SST WRS (Stream 2) adjusted for any loss of material to the FRPS Tanks Off-Gas stream (Stream 3). Nominal process throughput will be 6,000 gallons per day of undiluted tank waste at a 100% TOE. With a 1:1 dilution and 67% TOE, 8,040 gallons of slurry will be fed to the dryers per day (Design Inputs #20). A volume of 1,500 gallons of water was added for the initial charge to the WPU&SE, specified in the procurement specification document (Reference #7).

Stream 5 – FRPS Flush Water

The FRPS tanks are assumed to be flushed twice during the life of the operation with a volume equal to 25% of each tanks volume (Assumption #14). The first flush will occur before transferring the CH-TRUM WPU&SE to a new tank farm and the second flush will occur at the

end of operation. For the individual tank calculations, this flush water was added to the Tank B-204 and T-111 Process worksheets.

Streams 6 – Flowability Agent Addition to BMHS

A Flowability Agent (Vermiculite) will be used to establish a bed for dryer startup purposes. Stream 6 represents the receipt of material into the Bulk Material Handling System (BMHS) delivery vessels. The mass of Flowability Agent required to build this dryer bed was determined by dryer testing (Reference #3). Water content of the Flowability Agent was assumed to be negligible (Assumption #8).

Stream 7 – Scouring Agent Addition to BMHS

A Scouring Agent (Sand) is used as an additive during steady-state operation to provide a scouring action to reduce buildup on heating surfaces in the dryer. Stream 7 represents the receipt of material into the Bulk Material Handling System (BMHS) delivery vessels. The quantity of sand added is based on a dry end-point with 10 wt% water and 10 wt% scouring agent (Design Input #10). Water content of the Scouring Agent was assumed to be negligible (Assumption #8).

Stream 8 – Additive Feed to DWS Dryers

Stream 7 represents the feed stream from the BMHS delivery vessels into the dryer. The total quantity of dry material addition is the sum of Streams 6 and 7.

Stream 9 – DWS Flush

This is not a normal process, but is on the Process Flow Diagram to show the capability of the DWS dryers to be flushed if needed.

Intermediate Stage Dried Waste

This is not a process stream. The Intermediate Stage Dried Waste is the waste composition in the dryer when 10 wt% water remains in the dryer product.

Stream 10 – DWS ISO-Containers Air Inlet

Each dryer ISO-container is equipped with a HEPA breather filter. These air inlets are designed to allow an air sweep (650 scfm) through each compartment for heating or cooling. Stream 10 represents the sum of both of the dryer ISO-containers. The amount of water in this inlet stream is calculated from a relative humidity of 55% (Design Input #16) and an average temperature of 45°F (Design Input #15).

Stream 11 – Dried Waste Feed to WPS

Stream 11 is comprised of the dried product in the dryer. This combined material is fed to the Waste Packaging System (WPS) as a batch process. Using an estimated dried waste density of 1.60 kg/L (Assumption #3) allowed for a total volume of dryer product to be calculated.

Stream 12 – Sorbent Addition

Sorbent (Vermiculite) is added to each drum to absorb any condensate that forms (Criterion #12). The amount of sorbent added to a drum is based on the waterholding capacity of Vermiculite, which is 0.455 kg of Vermiculite per kg of water (Design Input #23). For this calculation, it was assumed that twice the amount of required sorbent is added to each drum (Assumption #10).

Stream 13 – WPS ISO-Container Air Inlet

The WPS is equipped with a HEPA-filtered air inlet. The unit is vented to maintain a slight negative pressure relative to atmospheric to minimize the potential of an uncontrolled release of material to the environment. This is accomplished via an air sweep in the same manner as for the FRPS Tanks. Air flows through the WPS at an assumed rate of 325 scfm (Design Input #14). The water content of this inlet is calculated using a relative humidity of 55% (Design Input #16) and a temperature of 45°F (Design Input #15). It is assumed that any water that accompanies the inlet air passes through the OGTS and out the stack (Assumption #11).

Stream 14 – Filled Waste Packages to Storage

Stream 14 represents the filled waste packages being removed from the WPS. This is a batch operation dependent on the rate at which the dryer can process waste for loading into the waste package. Criteria used to evaluate and load each waste package include net weight of waste, TRU content, FGE Pu-239, PE-Ci, DE-Ci and decay heat. TRU waste is defined as waste that contains alpha-emitting transuranic elements with half-lives greater than 20 years whose combined activity level is at least 100 nanocuries per gram of waste at the time of assay. The transuranic elements in the waste include Am-241, Am-243, Np-237, Pu-238, Pu-239, Pu-240, Pu-242, Cm-243 (Reference #14, Table A-2). Pu-241 is not included as it is a beta emitter and the half life is below 20 years. The minimum TRU content of 100 nCi TRU per gram of waste must be met for disposal at the Waste Isolation Pilot Plant (Criterion #8).

Stream 15 – DWS Dryers Off-gas

The dryer unit is operated nominally at 140°F under an approximate 26 inch Hg vacuum to evaporate water and dry the waste. The dryer drum wall will be operated at 250°F using a steam jacket. Volatile and some semi-volatile species remaining after waste retrieval and mixing in the FRPS are expected to be evolved. Other species in the waste may become entrained in the off-gas stream as a result of the mixing action inside of the dryer. The net loss of solids to the off-gas stream leaving the dryer is minimized by the use of a filter at the off-gas outlet of the dryer. This filter is periodically cleaned by blowing the material back into the dryer and an occasional water rinsing.

Due to similarities of function between the CH-TRUM WPU&SE dryer unit and the 242-A Evaporator used on the Hanford site, the partitioning of waste species into the off-gas from the dryer unit are modeled after the 242-A Evaporator. The retention factors from the 242-A evaporator data (Appendix L) were adjusted to account for chemicals and their isotopes having the same value. For those species that are not addressed in the 242-A Evaporator data, a partitioning factor of 8.0E+06 (mass that remains in the dryer divided by the mass that exits) is applied (Assumption #6). Refinement of the partitioning factors will be revised as data for the dryer are obtained.

Stream 16 – DWS ISO-Containers Off-Gas

Each dryer ISO-container is maintained at a slight negative pressure relative to atmospheric to provide secondary confinement. This is accomplished via an air sweep in the same manner as for the FRPS tanks. The air sweep also provides a mechanism for heating and cooling the compartment. This calculation assumes that the waste does not come into contact with the air sweep, so no waste is released into Stream 16. The water content of this stream is calculated by using an absolute humidity of 0.0035 and a temperature of 100°F.

Stream 17 – DWS Condensers Off-gas

The dryer off-gas stream is passed through a condenser to remove condensable species, mainly water, from the stream. Stream 17 represents the remaining species that are carried to the OGTS. The specific operating parameters of the condenser will dictate the partitioning of the inlet stream between the gas phase and liquid phase. The condenser unit is assumed to have the same retention factors as the 242-A evaporator (Assumption #6). Water lost into this stream is calculated using a relative humidity of 100% and a temperature of 100°F (Design Input #13). Each condenser is expected to have an off-gas flow rate of 150 scfm (Design Input #14).

Streams 18 and 19 – Liquid Effluent Feed to LES

Stream 18 represents the liquid effluent from the dryers that is collected in liquid effluent hold tanks as it is generated. Stream 19 represents batches of the liquid effluent sent from the liquid effluent hold tanks to one of two Liquid Effluent System (LES) Storage Tanks. Compositionally, the streams are the same and are based on the difference between the DWS Dryers Off-Gas (Stream 15) and the DWS Condensers Off-Gas (Stream 17).

Stream 20 – LES Storage Tanks Air Inlet

Both of the LES Storage Tanks are equipped with a HEPA-filtered air inlet, which provides a filtered exit path in the event of pressurization of the tank when liquid effluent is added. Stream 20 represents the sum of the streams into each individual tank. For this calculation, it was assumed that the air flow rate was zero.

Stream 21 – LES Flush Water

The LES Storage Tanks are assumed to be flushed twice during the life of the operation with a volume of 5,000 gallons of filtered water per tank per flush (Assumption #13). The first flush will occur before transferring the CH-TRUM WPU&SE to a new tank farm and the second flush

will occur at the end of operation. For the individual tank calculations, this flush water was added to the Tank B-204 and T-111 Process worksheets.

Stream 22 – WPS ISO-Container Off-gas

The WPS is maintained at a slight negative pressure relative to atmospheric to minimize the potential of an uncontrolled release of material to the environment. This is accomplished via an air sweep in the same manner as for the waste receipt tanks. A release factor of 5.98×10^{-9} (mass/m^3 in air)/ (mass/m^3 in liquid) from the Waste Treatment Plant HLW feed blending vessel (Reference #2) has been used to estimate the quantity of waste entrained in the WPS off-gas (Assumption #7). The total mass of waste lost to the off-gas is calculated by multiplying the release factor, the air flow rate through the unit, the density of the waste, and the total number of processing days. The amount of water lost to the off-gas is calculated using a relative humidity of 50% and a temperature of 75°F (Design Inputs #13).

Stream 23 – OGTS Inlet

Stream 23 represents the sum of the FRPS Tanks Off-Gas (Stream 3), the DWS Condensers Off-Gas (Stream 17), the WPS ISO-Container Off-Gas (Stream 22) and the DWS ISO-Containers Off-Gas (Stream 16).

Stream 24 – Stack Discharge

Off-gas from the CH-TRUM WPU&SE process goes through the OGTS to filter out contaminants before it is released to the atmosphere. Stream 24 is the filtered output from this system. For the purpose of showing the worst case scenario, it was assumed that 100% of the off-gas goes out through the Stack Discharge (i.e., no contaminants are filtered out) [Assumption #15].

Stream 25 – Loaded HEPA Filters

This stream represents the material collected on the HEPA filters located in the OGTS. These HEPA filters are treated as Secondary Solid Waste. Per the assumption for Stream 24, no contaminants are filtered out.

Stream 26 – OGTS Drain

Stream 23 is a periodic stream either from the addition of water into the OGTS or from liquid effluent generated and collected in the system. The stream is expected to be collected into a collection drum for disposal at ETF. Due to the assumption that all material entering the OGTS Inlet goes out with the Stack Discharge (Assumption #15), there is no entrained species in this stream.

Stream 27 – Pre-Filtered Liquid Effluent

Stream 27 is the sum of the Liquid Effluent Feed to the LES Storage Tanks (Stream 18/19) and the LES Flush Water (Stream 21). Liquid effluent is passed through a 5-micron filter to ensure that the stream specification criteria are met.

Stream 28 – Filtered Liquid Effluent

After passing through the 5-micron filter, liquid effluent is either sent to the SST WRS via an over-ground transfer line or loaded into a truck for disposition at the ETF. In either case, the composition of the stream must meet ETF acceptance criteria. For both destinations, Stream 25 is a batch operation.

Stream 28a – Filtered Liquid Effluent to ETF

This stream is the liquid effluent that is loaded into a truck for disposition at the ETF. The amount of liquid effluent is found by the total liquid effluent (Stream 28) minus the liquid effluent that is recycled to the WRS (Stream 28b).

Stream 28b – Filtered Liquid Effluent to WRS

This stream is the liquid effluent that is sent to the SST WRS via an over-ground transfer line. The amount of liquid effluent sent back to WRS for recycle is a percentage of the total liquid effluent (Stream 28). This percentage equates to 50% of the total dilution water used for retrieval (Assumption #20).

Stream 29 – Loaded 5-Micron Filters

The nature of the CH-TRUM WPU&SE process for dewatering and collecting the evolved water will minimize the potential for both dissolved and undissolved solid material in the liquid effluent, and therefore, the quantity of material collected by the 5-micron filter is expected to be negligible (Assumption #16).

Stream 30 – Secondary Solid Waste

Secondary Solid Waste will be disposed of in 55-gallon drums. It is estimated that the WPS will produce one 55-gallon drum of Secondary Solid Waste per day (Assumption #12). Secondary Solid Waste will include disposable gloves, plastic and tape for covering drum surfaces to prevent contamination, and wipes for survey and decontamination.

7.7 DRY BED ANALYSIS

Using SST Data (for the weight % water and total mass of tank waste), Tank Process Data (for the Stream 14 concentrations of transuranics, 137-Cs, and 90-Sr), and Assumptions and Criteria (for the dryer target fill, the dryer dried waste density, the initial addition of flowability agent into the dryer, the drum weight limit, and the wt% water in the dried product) a complete model for the dry bed operating method was created (see Appendix C for the Microsoft Excel® worksheets). In this model, six basic sections were used to simulate the process.

The first section is the dryer content. The dryer content has columns for solids, water, flowability agent, and scouring agent. These values are determined by taking the content in the dryer prior to dumping a drum minus the mass removed after dumping to the drum.

The second section is the dryer inputs. The dryer inputs section has columns for solids, water, cumulative total waste (solids plus water), feed tank number, flowability agent, and scouring agent. The flowability agent is only initially added to the dryer for dry bed startup. The scouring agent is then added in batches of 10% of the mass of a drum after the flowability agent has become significantly diluted. The solids added are determined by calculating all non-water mass lost to the drum dump minus the addition of flowability agent and scouring agent. The water added is then calculated using the solids added and the wt% water of the tank being pumped into the dryer. The cumulative total waste is the cumulative sum of all the solids and water waste added to the dryer. Finally, the "feed from" column calls out the number for the tank that is feeding the dryer.

The third section is the mass in dryer after inputs. These columns are simply a sum of the previous two sections for the solids, water, flowability agent, and scouring agent content. In addition to these totals is the approximate mass in the dryer from each tank. This is broken down into primary and secondary feeds. These values are determined using the cumulative total waste column in the previous section.

The fourth section is the drying section. The drying section has one column for condensate. This is the water evaporated from the dryer to achieve the desired 10 wt% dry product.

The fifth section is the dried product. These are the mass values and weight percents of the solids, water, flowability agent, and scouring agent left in the dryer after drying. The solids, flowability agents, and scouring agents are simply equal to their values prior to drying. The water is equal to its pre-dried mass minus the condensate. Subsequently, this is also the desired 10 wt% of the dryer mass which corresponds to approximately 1,915 kg.

The sixth, and final section, is the drum breakdown. This includes the composition of solids, water, flowability agent, and scouring agent in the drums. The mass of waste solids in the drum, as well as the tank from which the solids came, were used to determine the TRU content, FGE Pu-239, DE-Ci, PE-Ci, Decay Heat, 137-Cs content, and 90-Sr content. The TRU, 137-Cs, and 90-Sr contents were calculated by taking the solid mass in the drum times the respective tank concentrations to determine total curies in the drum. These total curies were then divided by the total mass in the drum (281 kg) to calculate the drum concentration. Similarly, the FGE Pu-239, DE-Ci, PE-Ci, and Decay Heat were calculated by summing the respective tank process data for each analyte, dividing by the respective total tank mass, and multiplying by solids mass in the drum.

8.0 USE OF COMPUTER SOFTWARE

This calculation was generated completely using an Excel® spreadsheet. The spreadsheet name is RPP-21970 Spreadsheet.xls. The spreadsheet owner is Ketra Clark. The spreadsheet is located on a CD and on the DMJM Technology file management system (E-Change). The calculation spreadsheet verification was performed in accordance with TFC-ENG-CHEM-D-33. The spreadsheet verification form number is SVF-237.

Excel® is a registered trademark of the Microsoft Corporation, Redmond, Washington.

9.0 RESULTS

This section describes three sets of results. Section 5.1 discusses the 11-Tank Average calculation, Section 5.2 discusses the Individual Tank calculation, and Section 5.3 discusses the Dry Bed Analysis calculation.

9.1 11-TANK AVERAGE

Based on 100% TOE, a nominal design feed rate of 6,000 gallons of undiluted waste per day and 1.4 million gallons of undiluted CH-TRUM waste, the total number of processing days is estimated at 235.5. This number was used to calculate the average nominal mass throughput per hour, day, and year, as well as the transuranic waste throughput per hour, for each stream. These results for the waste processing streams are shown on Table 9-1 and Table 9-2, along with the specific gravity, total mass, and total volume. Streams 25 (Loaded HEPA Filters) and 29 (Loaded 5-Micron Filters) are not shown because they are assumed to be zero to provide a conservative estimate for the Stack Discharge (Stream 24) and Filtered Liquid Effluent (Stream 28). Also note that mass for the gas streams do not contain a mass for air. This allows for a clearer illustration of how much waste is conveyed to the Off-Gas Treatment System (OGTS).

The duration of the waste processing and packaging campaign would be equal to the total number of processing days if the system was 100% efficient and 100% utilized. Using a total operating efficiency of 67% (Design Input #20) to account for equipment transfer, downtime, maintenance, or failure, the time to process the waste increases to 351.5 days.

While filling a drum, the volume limit is reached before the WIPP 55-gallon drum weight limit of 1,000 lbs. Therefore, a new waste weight limit of 620 lbs for each drum was calculated using the internal volume of the drum, the 85% volume fill limit, and a dried product density of 1.60 kg/L. Using this weight limit, a total of 7,548 drums will be required to process all of the dried waste/additive.

The calculated average decay heat from an drum is 0.02 watt. The average dose equivalent curie per drum is 0.16 DE-Ci. The calculated average Pu-239 FGE per drum is 7.3. The packaged product has an average of 586 nCi of TRU per gram of packaged waste.

Stream 28 represents the filtered liquid effluent that is to either be returned to the SST WRS or is to be sent to ETF for treatment and disposition. Per the CH-TRUM procurement specification (Reference #7), the filtered liquid effluent must meet the ETF waste acceptance criteria. ETF maintains an Ecology-approved influent constituents list defining all constituents and maximum concentrations acceptable for treatment in the ETF (Reference #18). This list is provided in Appendix O. A comparison of the concentration of analytes in Stream 28 to this list is provided in Appendix F. Uranium-234 is the only constituent that is estimated to be over the limit set in the waste acceptance criteria.

9.2 INDIVIDUAL TANK CALCULATION

Results for the main waste processing parameters for each tank are shown on Tables 9-3 through 9-8. The CH-TRUM Tanks will be retrieved sequentially, making it critical to compare each individual tank's results with specified limits for certain quantities (e.g. FGE per drum).

The calculated Pu-239 FGE per drum ranges from 1.1 for Tank T-110 to 16.9 for Tank T-111. Tank T-111, therefore, has a smaller allowed standard deviation (540%) in comparison with the other ten tanks. The PE-Ci per drum ranges from 0.07 to 0.49. The DE-Ci per drum also ranges between 0.07 to 0.49.

The transuranic content of the FRPS Slurry Feed (Stream 2), the Filled Waste Packages (Stream 14), and the Liquid Effluent (Streams 18/19) for each of the tanks are shown in Table 9-5. TRU concentrations range between 38 to 455 nCi/g in the feed, 249 to 1,734 nCi/g in the drum, and 3.47×10^{-5} to 1.72×10^{-4} nCi/g in the liquid effluent.

Tables 9-6 through 9-8 show the maximum and minimum concentrations of Cs-137 and Sr-90 in Steam 2 (FRPS Slurry Feed), Stream 14 (Filled Waste Packages) and Streams 18/19 (Liquid Effluent). Inventory uncertainties from the BBI data were included in these concentrations to give a worst case scenario which are used for shielding analysis. The uncertainty estimates are relative standard deviations presented as fractions. Inventory uncertainty is calculated by combining volume, concentration, and density uncertainties.

9.3 DRY BED ANALYSIS

Tabulated results for the dry bed operating scenario are shown in Appendix C-1 for B-Farm and Appendix C-2 for T-Farm. In summary, 876 complete drums will be filled for the B-Farm tanks and 6,731 complete drums will be filled for the T-Farm tanks. In these 7,607 drums, there will be 1,707,642 kg of solid waste, 213,757 kg of water, 9,820 kg of Flowability Agent, and 206,349 kg of Scouring Agent. For both farms, the flowability agent, which is only initial added to the dryer, makes up less than 5% of the dryer mass after dumping 112 drums, and the scouring agent in the dryer is within 1% of its steady-state concentration after dumping 216 drums.

Figures 9-1 and 9-2 show graphs of TRU content versus flowability agent and scouring agent for the B-Farm and T-Farm respectively. Even with dilution in the drum from the flowability agent and the scouring agent, the TRU content for processing B-Farm tanks never drops below 660 nCi/g, and the TRU content for processing T-Farm tanks never drops below 246 nCi/g. These results assume a process order of B-201, B-202, B-203, and B-204 for the B-Farm tanks and T-201, T-202, T-203, T-204, T-110, T-111, and T-104 for the T-Farm tanks (Assumption #2). Given the application of the flowability agent at the beginning of the drying operation, the TRU content in the initial drums is reduced significantly. This reduction is approximately 20% of the steady-state operation, in which the presence of flowability agent is negligible and the scouring agent content is 10%. If a different processing order was used and the tank (T-110) with the lowest TRU concentration (246 nCi/g) was processed first this would still result in an acceptable drum TRU content of approximately 197 nCi/g (= 0.8 * 246 nCi/g) as compared to a steady-state content of 246 nCi/g.

For the B-Farm, the DE-Ci values range from about 0.18 to 0.49, the PE-Ci values range from about 0.19 to 0.50, and the Decay Heat values are less than 0.02 Watts. For the T-Farm, the DE-Ci values range from about 0.07 to 0.54, the PE-Ci values range from about 0.07 to 0.54, and the Decay Heat values are less than 0.03 Watts.

Figure 9-1. Graph of B-Farm 55-Gallon Drum TRU Concentration and Dryer Content of Additives.

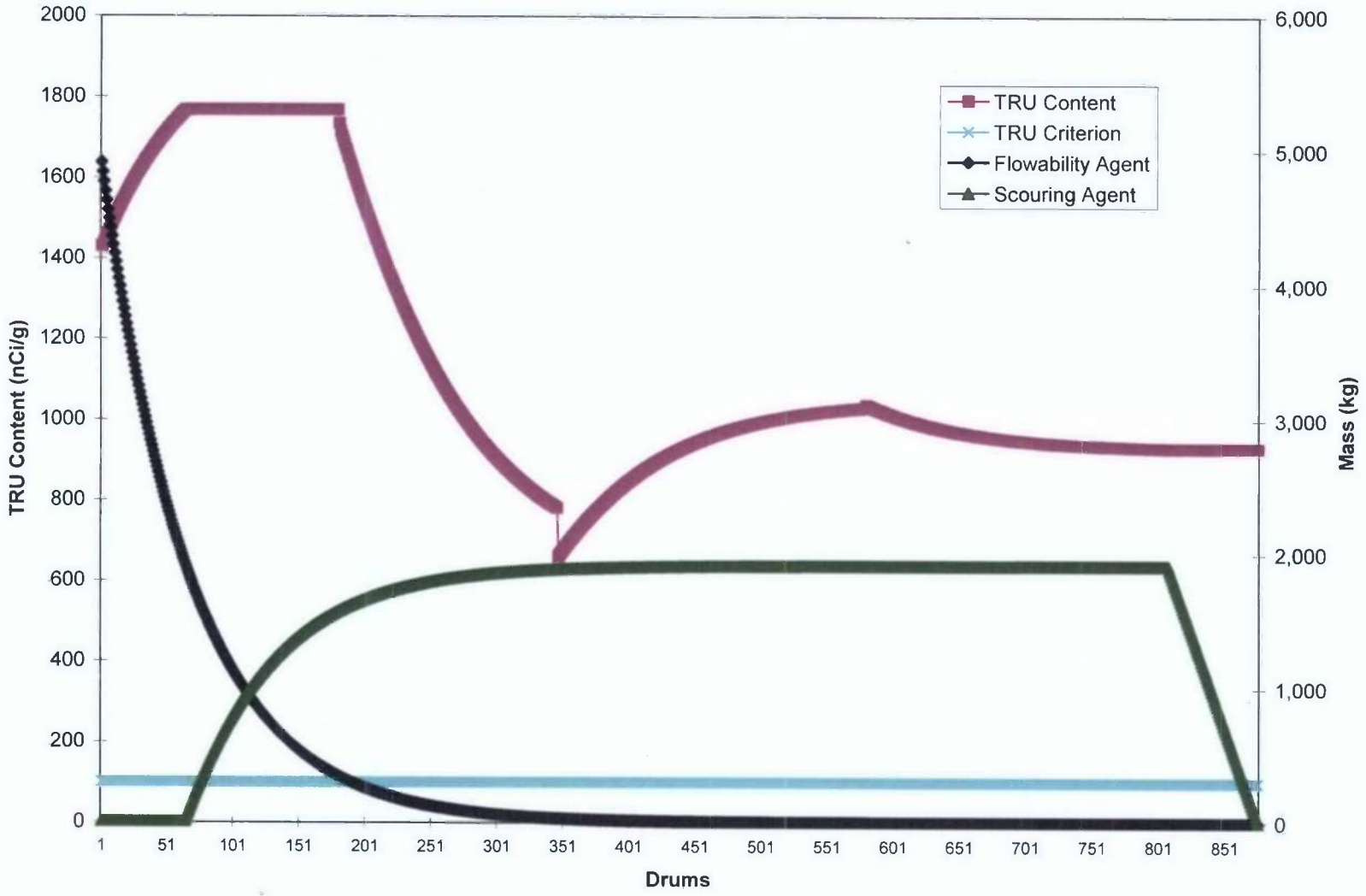

Figure 9-2. Graph of T-Farm 55-Gallon Drum TRU Concentration and Dryer Content of Additives.

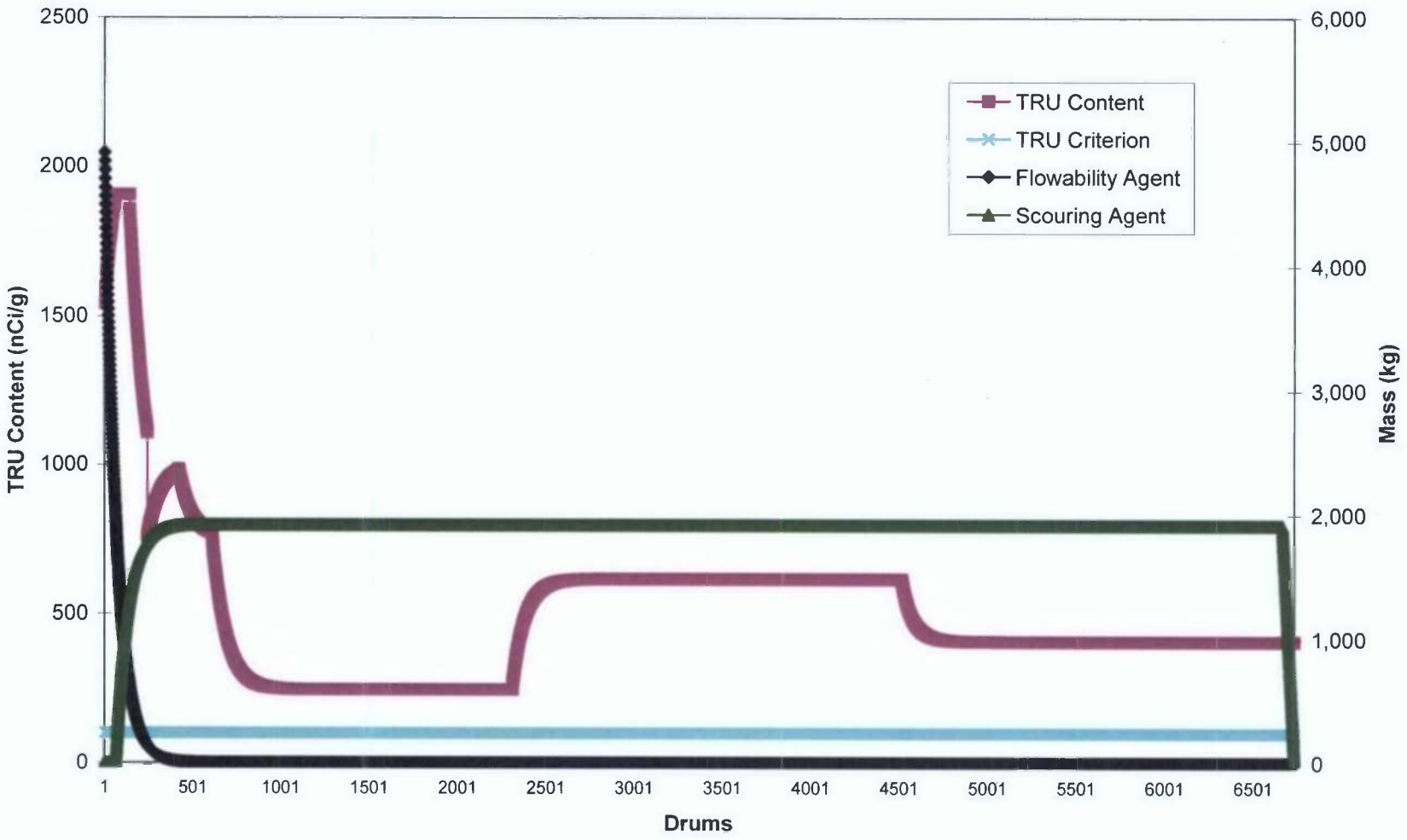

Table 9-1. Material Balance Data by Stream Number – Main Process Streams. (Sheet 1 of 2)

Stream Number		2	4	5	6	7	8	9	11	12	14
Stream Name	**Units**	**WRS SLURRY FEED TO FRPS TANKS**	**SLURRY FEED TO DWS DRYERS**	**FRPS FLUSH WATER**	**FLOW AGENT ADDITION TO BMHS**	**SCOUR AGENT ADDITION TO BMHS**	**ADDITIVE FEED TO DWS DRYERS**	**DWS FLUSH**	**DRIED WASTE FEED TO WPS**	**SORBENT ADDITION TO WPS**	**FILLED WASTE PACKAGE**
Phase		SLURRY	SLURRY	LIQUID	SOLID	SOLID	SOLID	SOLID	LIQUID	LIQUID	LIQUID
Nominal Through-put	lb/hr	4.68E+03	4.69E+03	2.66E+01	n/a	n/a	n/a	n/a	8.36E+02	n/a	8.27E+02
**Note*	lb/day	1.12E+05	1.12E+05	6.38E+02	n/a	n/a	n/a	n/a	2.01E+04	n/a	1.99E+04
	lb/year	2.65E+07	2.65E+07	1.50E+05	2.16E+04	4.73E+05	4.94E+05	0	4.73E+06	n/a	4.68E+06
Specific gravity		1.12	1.12	1.00	0.13	1.35	n/a	1.00	1.60	0.13	1.60
Transuranic	nCi/g	1.03E+02	1.03E+02	n/a	n/a	n/a	n/a	n/a	5.79E+02	n/a	5.86E+02
**Note*	Ci/hr	2.20E-01	2.20E-01	n/a	n/a	n/a	n/a	n/a	2.20E-01	n/a	2.20E-01
Total Volume	gal	2.82E+06	2.82E+06	1.80E+04	n/a	n/a	n/a	0	3.54E+05	n/a	3.50E+05
Total Mass	lb	2.65E+07	2.65E+07	1.50E+05	2.16E+04	4.73E+05	4.94E+05	0	4.73E+06	1.62E+02	4.68E+06

*Note: 235.5 day year, continuous basis

Table 9-2. Material Balance Data by Stream Number – Main Process Streams. (Sheet 2 of 2)

Stream Number		18	19	21	26	27	28	28a	28b	30
Stream Name	Units	LIQUID EFFLUENT FEED TO HOLD TANKS	LIQUID EFFLUENT FEED TO LES STORAGE TANKS	LES FLUSH WATER	OGTS DRAIN	PRE-FILTERED LIQUID EFFLUENT	FILTERED LIQUID EFFLUENT	FILTERED LIQUID EFFLUENT TO ETF	FILTERED LIQUID EFFLUENT TO WRS	SOLID SECOND WASTE
Phase		LIQUID	LIQUID	LIQUID	LIQUID	LIQUID	LIQUID	LIQUID	LIQUID	SOLID
Nominal Through-put	lb/hr	3.91E+03	3.91E+03	2.95E+01	0	3.94E+03	3.94E+03	2.90E+03	1.04E+03	n/a
*Note	lb/day	9.37E+04	9.37E+04	7.09E+02	0	9.44E+04	9.44E+04	6.95E+04	2.50E+04	n/a
	lb/year	2.21E+07	2.21E+07	1.67E+05	0	2.22E+07	2.22E+07	1.64E+07	5.88E+06	n/a
Specific gravity		1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	n/a
Transuranic	nCi/g	4.86E-05	4.86E-05	n/a	n/a	4.82E-05	4.82E-05	4.82E-05	4.82E-05	n/a
*Note	Ci/hr	8.61E-08	8.61E-08	n/a	0	8.61E-08	8.61E-08	6.34E-08	2.28E-08	0
Total Volume	gal	2.65E+06	2.65E+06	2.00E+04	0	2.67E+06	2.67E+06	1.96E+06	7.05E+05	1.30E+04
Total Mass	lb	2.21E+07	2.21E+07	1.67E+05	0	2.22E+07	2.22E+07	1.64E+07	5.88E+06	n/a

*Note: 235.5 day year, continuous basis

Table 9-3. Material Balance Data by Stream Number – Gas Streams.

Stream Number		1	3	10	13	15	16	17	20	22	23	24
Stream Name	**Units**	**FRPS TANKS AIR INLET**	**FRPS TANKS OFF-GAS**	**DWS ISO-CONT AIR INLET**	**WPS ISO-CONT AIR INLET**	**DWS DRYERS OFF-GAS**	**DWS ISO-CONT OFF-GAS**	**DWS CONDENSER OFF-GAS**	**LES TANKS AIR INLET**	**WPS ISO-CONT. OFF-GAS**	**OGTS INLET**	**STACK DISCHARGE**
Phase		GAS	GAS	GAS	GAS	3.29E+03	2.18E+01	5.81E+01	0	GAS	GAS	GAS
Nominal Throughput	lb/hr	n/a	1.53E+00	n/a	n/a	3.96E+03	2.18E+01	5.81E+01	0	1.42E+01	9.57E+01	9.57E+01
**Note*	lb/day	n/a	3.66E+01	n/a	n/a	9.51E+04	5.23E+02	1.40E+03	0	3.41E+02	2.30E+03	2.30E+03
	lb/year	4.74E+03	8.62E+03	1.23E+05	30006	2.24E+07	1.23E+05	3.29E+05	0	8.03E+04	5.41E+05	5.41E+05
Volumetric Flow Rate	scfm	50	50	1300	325	n/a	1300	300	0	325	1975	1975
Total Mass	lb	4.74E+03	8.62E+03	1.23E+05	3.00E+04	2.24E+07	1.23E+05	3.29E+05	n/a	8.03E+04	5.41E+05	5.41E+05
Total Radionuclides	Ci	0	1.22E-03	n/a	n/a	8.15E-01	0	2.29E-04	0	3.72E-01	3.74E-01	3.74E-01

*Note: 235.5 day year, continuous basis

Table 9-4. Material Balance Data by Tank – Key Process Parameters (Sheet 1 of 2)

Stream Number	Existing Tank Waste				2	2	5	18
	Dry Mass MT	Sludge (gal)	Supernate (gal)	Total (gal)	wt% Water in Feed	Feed Volume includes 1:1 Dilution (gal)	Total FRPS Flush Water (gal)	Condensate Vol (gal)
B-201	54.5	30,923	0	30,923	79.4%	61,846	0	54,177
B-202	37.2	30,288	0	30,288	85.4%	60,576	0	56,834
B-203	52.4	50,193	530	50,724	87.5%	100,917	0	95,001
B-204	51.5	49,968	789	50,757	87.7%	100,725	9,000	104,044
T-201	45.9	28,721	2,099	30,820	82.3%	59,541	0	55,487
T-202	22.5	21,265	0	21,265	87.1%	42,530	0	40,561
T-203	39.7	36,187	0	36,187	86.9%	72,374	0	69,027
T-204	43.0	37,587	0	37,587	86.1%	75,175	0	69,595
T-110	382.3	359,302	756	360,058	87.5%	719,360	0	687,070
T-111	495.4	442,314	0	442,314	86.8%	884,629	9,000	841,774
T-104	483.0	322,300	0	322,300	82.7%	644,599	0	587,034
TOTAL	1,707	1,409,048	4,175	1,413,223		2,822,271	18,000	2,660,603

Table 9-5. Material Balance Data by Tank – Key Process Parameters (Sheet 2 of 2)

Stream Number	6	7	14	14		14	14
	Flowability Agent Vol (gal)	Scouring Agent Vol (gal)	Packaged mass (kg)	Packaged Volume [ρ= 1.6 kg/L] (gal)	Days Processing	Total Drums	Drums/Day
B-201	10,377	1,448	73,675	12,164	5.15	262.1	50.9
B-202	0	907	45,996	7,594	5.05	163.6	32.4
B-203	0	1,277	64,583	10,663	8.45	229.7	27.2
B-204	0	1,255	63,468	10,479	8.46	225.8	26.7
T-201	10,377	1,239	62,984	10,399	5.14	224.0	43.6
T-202	0	550	27,825	4,594	3.54	99.0	27.9
T-203	0	968	48,997	8,090	6.03	174.3	28.9
T-204	0	1,050	53,170	8,779	6.26	189.1	30.2
T-110	0	9,319	471,600	77,865	60.01	1,677.6	28.0
T-111	0	12,073	611,336	100,936	73.72	2,174.6	29.5
T-104	0	11,778	598,152	98,760	53.72	2,127.7	39.6
TOTAL	20,753	41,864	2,121,787	350,324	236	7,548	

Table 9-6. Material Balance Data by Tank – Key Process Parameters.

	FGE Pu-239 per Drum	Allowed STD in FGE Pu-239	DE-Ci per Drum	Decay Heat Watts per Drum	PE-Ci per Drum
B-201	7.1	1362%	0.46	0.02	0.46
B-202	2.9	3405%	0.17	0.02	0.18
B-203	3.9	2488%	0.29	0.01	0.30
B-204	4.6	2145%	0.26	0.02	0.26
T-201	7.5	1284%	0.49	0.02	0.49
T-202	3.6	2746%	0.21	0.01	0.21
T-203	3.7	2675%	0.28	0.01	0.28
T-204	2.8	3475%	0.21	0.01	0.21
T-110	1.1	9317%	0.07	0.00	0.07
T-111	16.9	540%	0.17	0.03	0.18
T-104	4.3	2275%	0.11	0.01	0.12

Table 9-7. Transuranic Content of Key Process Streams by Tank.

	TRU					
	Stream 2		Stream 14		Stream 18/19	
	Total Ci	nCi/g	Total Ci	nCi/g	Total Ci	nCi/g
B-201	1.20E+02	454.5	1.20E+02	1632.1	3.52E-05	1.72E-04
B-202	2.92E+01	114.9	2.92E+01	635.6	1.34E-05	6.23E-05
B-203	6.86E+01	163.9	6.86E+01	1061.6	2.32E-05	6.44E-05
B-204	5.94E+01	142.4	5.94E+01	936.3	1.65E-05	4.18E-05
T-201	1.09E+02	420.4	1.09E+02	1734.4	3.02E-05	1.44E-04
T-202	2.12E+01	120.8	2.12E+01	762.0	5.87E-06	3.82E-05
T-203	4.96E+01	163.0	4.96E+01	1011.4	1.71E-05	6.55E-05
T-204	4.05E+01	130.7	4.05E+01	762.4	1.35E-05	5.14E-05
T-110	1.17E+02	38.3	1.17E+02	248.7	9.03E-05	3.47E-05
T-111	3.81E+02	101.6	3.81E+02	623.4	1.57E-04	4.92E-05
T-104	2.46E+02	88.2	2.46E+02	411.8	8.48E-05	3.82E-05
Max	381	455	381	1,734	1.57E-04	1.72E-04
Min	21	38	21	249	5.87E-06	3.47E-05

Table 9-8. Cesium and Strontium Concentrations in Waste Feed (Stream 2).

	137-Cs					90-Sr				
	Total Ci	% Tot. Ci Unc.	Tot. Ci Unc. (±)	Total Ci w/Unc.	nCi/g	Total Ci	% Tot. Ci Unc.	Tot. Ci Unc. (±)	Total Ci w/ Unc.	nCi/g
B-201	18.7	0.225	4.2	23.0	86.8	235.7	0.259	61.0	296.7	1121.6
B-202	9.9	0.460	4.5	14.4	56.5	385.7	0.164	63.3	449.0	1764.1
B-203	1.7	0.083	0.1	1.9	4.4	15.3	0.087	1.3	16.7	39.8
B-204	5.9	1.003	5.9	11.9	28.4	277.4	14.455	4010.2	4287.6	10272.8
T-201	3.9	0.376	1.5	5.4	20.7	9.9	0.392	3.9	13.8	53.3
T-202	2.3	1.004	2.3	4.6	26.5	0.2	0.686	0.1	0.3	2.0
T-203	2.2	0.082	0.2	2.4	7.9	0.4	0.089	0.0	0.4	1.3
T-204	1.1	0.079	0.1	1.2	3.8	0.6	0.260	0.2	0.8	2.6
T-110	24.7	0.095	2.4	27.1	8.8	45.2	0.113	5.1	50.2	16.4
T-111	202.5	0.364	73.7	276.2	73.6	8138.0	0.216	1759.0	9897.1	2638.9
T-104	228.3	0.087	19.8	248.0	88.8	2983.4	0.102	302.9	3286.3	1176.3
Max	228.3	1.0	73.7	276.2	88.8	8138.0	14.5	4010.2	9897.1	10272.8
Min	1.101	0.079	0.087	1.188	3.831	0.203	0.087	0.032	0.342	1.286

Table 9-9. Cesium and Strontium Concentrations in Filled Waste Packages (Stream 14).

	137-Cs					90-Sr				
	Total Ci	% Tot. Ci Unc.	Tot. Ci Unc. (±)	Total Ci w/Unc.	nCi/g	Total Ci	% Tot. Ci Unc.	Tot. Ci Unc. (±)	Total Ci w/ Unc.	nCi/g
B-201	18.7	0.225	4.2	23.0	311.7	235.7	0.259	61.0	296.7	4027.2
B-202	9.9	0.460	4.5	14.4	312.9	385.7	0.164	63.3	449.0	9762.2
B-203	1.7	0.083	0.1	1.9	28.6	15.3	0.087	1.3	16.7	258.0
B-204	5.9	1.003	5.9	11.9	186.9	277.4	14.455	4010.1	4287.5	67553.9
T-201	3.9	0.376	1.5	5.4	85.4	9.9	0.392	3.9	13.8	219.7
T-202	2.3	1.004	2.3	4.6	166.9	0.2	0.686	0.1	0.3	12.3
T-203	2.2	0.082	0.2	2.4	49.2	0.4	0.089	0.0	0.4	8.0
T-204	1.1	0.079	0.1	1.2	22.3	0.6	0.260	0.2	0.8	15.3
T-110	24.7	0.095	2.4	27.1	57.4	45.2	0.113	5.1	50.2	106.5
T-111	202.5	0.364	73.7	276.2	451.8	8137.9	0.216	1759.0	9896.9	16188.9
T-104	228.3	0.087	19.8	248.0	414.7	2983.3	0.102	302.9	3286.3	5494.0
Max	228.3	1.0	73.7	276.2	451.8	8137.9	14.5	4010.1	9896.9	67553.9
Min	1.1	0.1	0.1	1.2	22.3	0.2	0.1	0.0	0.3	8.0

Table 9-10. Cesium and Strontium Concentrations in Dryer Liquid Effluent (Stream 18/19).

	137-Cs					90-Sr				
	Total Ci	% Tot. Ci Unc.	Tot. Ci Unc. (±)	Total Ci w/Unc.	nCi/g	Total Ci	% Tot. Ci Unc.	Tot. Ci Unc. (±)	Total Ci w/ Unc.	nCi/g
B-201	2.03E-07	0.225	4.58E-08	2.49E-07	1.21E-06	1.03E-03	0.259	2.66E-04	1.30E-03	6.32E-03
B-202	1.07E-07	0.460	4.91E-08	1.56E-07	7.25E-07	1.68E-03	0.164	2.76E-04	1.96E-03	9.11E-03
B-203	1.85E-08	0.083	1.54E-09	2.01E-08	5.58E-08	6.69E-05	0.087	5.85E-06	7.27E-05	2.02E-04
B-204	6.42E-08	1.003	6.44E-08	1.29E-07	3.27E-07	1.21E-03	14.455	1.75E-02	1.87E-02	4.75E-02
T-201	4.24E-08	0.376	1.60E-08	5.83E-08	2.78E-07	4.34E-05	0.392	1.70E-05	6.04E-05	2.88E-04
T-202	2.51E-08	1.004	2.52E-08	5.04E-08	3.28E-07	8.86E-07	0.686	6.08E-07	1.49E-06	9.73E-06
T-203	2.42E-08	0.082	1.99E-09	2.62E-08	1.00E-07	1.57E-06	0.089	1.40E-07	1.71E-06	6.54E-06
T-204	1.19E-08	0.079	9.42E-10	1.29E-08	4.89E-08	2.82E-06	0.260	7.33E-07	3.55E-06	1.35E-05
T-110	2.68E-07	0.095	2.55E-08	2.94E-07	1.13E-07	1.97E-04	0.113	2.22E-05	2.19E-04	8.43E-05
T-111	2.20E-06	0.364	7.99E-07	3.00E-06	9.40E-07	3.55E-02	0.216	7.68E-03	4.32E-02	1.36E-02
T-104	2.48E-06	0.087	2.14E-07	2.69E-06	1.21E-06	1.30E-02	0.102	1.32E-03	1.43E-02	6.46E-03
Max	2.48E-06	1.00E+00	7.99E-07	3.00E-06	1.21E-06	3.55E-02	1.45E+01	1.75E-02	4.32E-02	4.75E-02
Min	1.19E-08	7.89E-02	9.42E-10	1.29E-08	4.89E-08	8.86E-07	8.75E-02	1.40E-07	1.49E-06	6.54E-06

10.0 CONCLUSIONS

The total processing days at a nominal process throughput of 6,000 gpd at 100% TOE is 235.5 days. With the addition of a 67% TOE, the projected mission duration is calculated to be 351.5 days. This allows for periods of lower waste feed rates, and allows for additional down-time not accounted for. This also shows that there is a small amount of flexibility for accepting waste batches with more dilution water. Adding more dilution water reduces the undiluted waste throughput per day due to the processing limitations of the dryer.

TRU content, Pu-239 FGE activity, Pu-239 Equivalent (PE-Ci) activity, decay heat, and dose equivalent (DE-Ci) activity of the dried product were analyzed on an 11-tank average basis, an individual tank basis, and a dry bed drum-by-drum basis. These values were then compared to limits called out in the waste acceptance criteria for the dried product.

The dried product meets the WIPP waste acceptance criteria for each drum by containing more than 100 nCi/g of alpha-emitting TRU isotopes (Criterion #8), with an average of 586 nCi/g of packaged waste, a tank range average of 249 to 1,735 nCi/g and a drum range of 246 to 1,905 nCi/g.

The Pu-239 FGE was calculated for a comparison to its drum limit. For a drum, the Pu-239 FGE plus 2 times the standard deviation must be less than the 200 gram limit (Criterion #7). The dried product meets this criterion by averaging 7.3 Pu-239 FGE per drum on an overall average basis, a range of 1.1 to 16.9 Pu-239 FGE on an individual tank basis, and a range of 1.05 to 16.7 for the drum-by-drum approach.

The dried product meets the WIPP waste acceptance criteria for each drum by containing less than or equal to 80 PE-Ci (Criterion #9). The eleven tank average estimates the PE-Ci per drum at 0.17. The individual tank basis estimates a range between 0.07 and 0.49 PE-Ci per drum. The drum-by-drum approach estimates a range between 0.07 and 0.54 PE-Ci per drum.

The average dose equivalent curie per drum was calculated for comparison to the WIPP waste acceptance criteria limit of 82.5 DE-Ci (Criterion #11). The individual drums successfully meet this DE-Ci limit with an average value of 0.16 DE-Ci, an individual tank range average of 0.07 to 0.49 DE-Ci and a drum-by-drum approach range of 0.07 and 0.54 DE-Ci.

The dried product meets the Hanford Site solid waste acceptance criteria (Reference #14) for each drum by containing less than the decay heat limit of 0.1 watt/ft^3 or 6.63 watts (Criterion #10). The eleven tank average estimates the decay heat per drum as well below this limit at 0.02 Watts. The individual tank basis estimates a maximum of 0.03 Watts per drum. The drum-by-drum approach also estimates a maximum of 0.03 Watts per drum.

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12.0 APPENDICES

Appendix	Title
A	Process Flow Diagram: Drawing 20843.P001, Rev. 1
B	Calculation Worksheet: Process Baseline
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J	Excerpt from *Integrated Emissions Baseline Report*, RPT-W375-ES00001, Rev. 1
K	Excerpt from Hanford Site Solid Waste Acceptance Criteria, HNF-EP-0063, Rev. 10, Table A-1 and Table B-1
L	Email from Ivan Papp to John Garfield, *"242-A Evaporator Partitioning"* with excerpt from attached file 242-A Evaporator Partitioning.xls
M	Excerpt from *Tank Farm Contractor Operation and Utilization Plan*, HNF-SD-WM-SP-012, Rev. 4
N	Excerpt from Contact-Handled Transuranic Waste Acceptance Criteria for the Waste Isolation Pilot Plant, DOE/WIPP-02-3122, Rev. 0.1, Table B-1
O	Email from Mark Bowman to Paul Day, *"Waste Acceptance Document, ST4500 Constituents and LERF Rad Limits"* with attached file List of Constituents.xls
P	Calculation Review Checklists

APPENDIX A

PROCESS FLOW DIAGRAM

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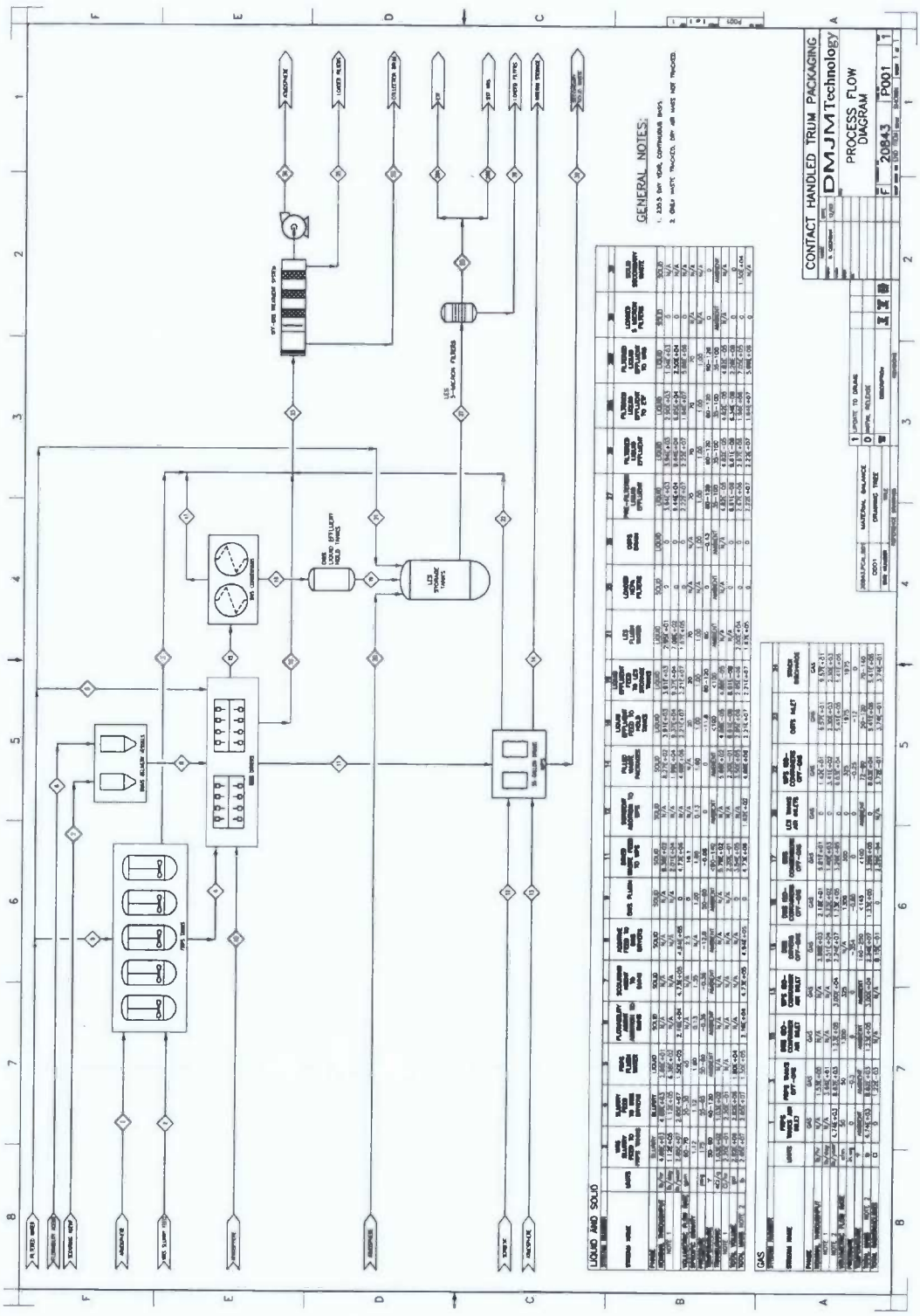
CONTACT HANDLED TRUM PACKAGING
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PROCESS FLOW DIAGRAM
F 20843 P001
GENERAL NOTES:
LIQUID AND SOLID
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ENGINEERING DATA TRANSMITTAL

1. EDT 820962

1A. Page 1 of 1

2. To: (Receiving Organization)
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3. From: (Originating Organization)
Closure Projects

4. Related EDT No.: N/A

7. Purchase Order No.: N/A

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TRU Project

6. Design Authority/Resp. Engr./Design Agent:
R.L. Brown

9. Equip./Component No.: N/A

10. System/Bldg./Facility: N/A

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1	RPP-21475		0	Contact Handled - Transuranic Waste High Dose Rate Issue Resolution Study	N/A	1	1	

16. KEY

Approval Designator (F)	Reason for Transmittal (G)	Disposition (H) & (I)
See TFC-ESHQ-Q-INSP-C-05	1. Approval 2. Review 3. Post-Review	1. Approved 2. Approved w/comment 3. Reviewed no comment 4. Reviewed w/comment 5. Disapproved

17. SIGNATURE/DISTRIBUTION

(G) Reason	(H) Disp.	(J) Name	(K) Signature	(L) Date	(M) MSIN	(G) Reason	(H) Disp.	(J) Name	(K) Signature	(L) Date	(M) MSIN
		Design Auth.				1	1	R.M. Pierson	R.M. Pierson	07/15/04	S7-83
1	1	Resp. Engr. R.L. Brown	RLBrown	7-8-04	S7-75	1	1	W.E. Bryan	W.E. Bryan	8/2/04	H6-19
1	1	Resp. Mgr. J.W. Hobbs	J.W. Hobbs	7-12-04	S7-65	1	1	M.E. Johnson	M.E. Johnson	8/5/04	H6-19
		QA									
		Safety									
		Env.									
		Design Agent									
		M.R. Landon									

18. Signature of EDT Originator: Matt Landon 7/7/04 Date

19. DOE APPROVAL (if required) Ctrl. No. N/A

20. Design Auth./Resp. Engr./Resp. Mgr.: John W. Hobbs 8/2/04 Date

BD-7400-172.2 (08/03)

DISTRIBUTION SHEET

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Distribution	TRU Project	Date 07/07/04
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Name	MSIN	Text With All Attach.	Text Only	Attach./ Appendix Only	EDT/ECN Only
G.L. Amidan	H6-16	X			
D.G. Baide	S7-70	X			
K.D. Boomer	R1-44	X			
P.M. Branson	S7-12	X			
R.L. Brown	S7-75	X			
W.E. Bryan	H6-19	X			
D.D. Carson	H6-19	X			
R.E. Heath	T4-08	X			
G.P. Janicek	S7-12	X			
M.J. Jennings	H6-03	X			
M.E. Johnson	R1-44	X			
S.D. Kozlowski	S4-44	X			
A.R. Tedeschi	H6-19	X			
W.T. Thompson	S7-70	X			
T.A. Wooley	R1-51	X			
M.R. Landon	H6-19	X			
M.J. Holm	H6-19	X			
C.R. Reichmuth	S7-20	X			
D.S. Rewinkel	R3-86	X			
F.R. Miera	H6-03	X			
J.W. Hobbs	S7-75	X			
R.M. Pierson	S7-83	X			
Central Files	B1-07	X			

Contact Handled - Transuranic Waste High Dose Rate Issue Resolution Study

Richland, WA 99352
U.S. Department of Energy Contract DE-AC27-99RL14047

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Key Words:
TRU, Waste Packaging Unit, T-111, Contact Handled, Transuranic Waste, Dose Rate Study

Abstract:
This study estimates dose rates for different waste packing configurations for the TRU Project and alternatives to deal with potentially higher dose TRU waste.

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CONTACT HANDLED - TRANSURANIC WASTE HIGH DOSE RATE ISSUE RESOLUTION STUDY

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CH2M HILL Hanford Group, Inc.

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EXCECUTIVE SUMMARY

In FY 2004 CH2M HILL issued a multi-year contract to design and deploy a Contact Handled Transuranic Mixed (CH-TRUM) Waste Packaging System (WPS) to supplement the non-High Level Waste (HLW) treatment capacity provided by the Waste Treatment Plant (WTP). Ten Single Shell Tanks (SST): B-201, B-202, B-203, B-204, T-201, T-202, T-203, T-204, T-110, and T-111 have been identified as candidate CH-TRUM waste tanks. Waste from these tanks will be retrieved, packaged, and ultimately disposed of at the Carlsbad Waste Isolation Pilot Plant (WIPP). The goal is to package and prepare for disposition 1.1 million gallons of CH-TRUM waste by September 2006.

The CH-TRUM WPS uses dryer technology to dewater retrieved tank waste. The dewatering process employed by the CH-TRUM WPS concentrates Cs-137 in the final dryer product and results in Cs-137 concentrations that are higher than those reported for the in-tank waste. In addition, an analysis of composite core samples obtained from Tank T-111 identified a surface layer (~ top 40 inches) with specific activity higher than the mean concentration reported in the Tank Waste Information Netware System (TWINS).

The WIPP acceptance criteria for CH-TRUM waste includes requirements that the external radiation dose rate of payload containers (e.g., SWB, 55-gallon drum) and packaged payload assemblies (i.e., TRUPACT-II) be less than or equal to 200 mrem/hour at contact, and that the external dose rate of payload assemblies be less than or equal to 10 mrem/hour at 2 meters (DOE/WIPP-02-3122). In addition, the Hanford Site Solid Waste Acceptance Criteria limits the external dose rate on payload containers to less than or equal to 100 mrem/hour at 30 centimeters (HNF-EP-0063)

The current technical baseline for the WPS specifies that dewatered waste will be packaged in Standard Waste Boxes (SWB). However, over the past several months results of process development testing, vendor design reviews, and analyses conducted by the Pacific Northwest National Laboratory (PNNL) have indicated that modifications to the CH-TRUM WPS design and/or operational approach may be necessary to mitigate potential high radiation dose rate issues associated with packaged CH-TRUM waste. Uncertainties associated with final packaged waste densities, increased specific activities as a consequence of the drying process, tank-specific contributions from Bremsstrahlung radiation, and increased Cs-137 concentrations in T-111 confirmed the need for a detailed dose rate evaluation.

Three tanks: B-201, B-202, and T-111 are addressed in this study. These tanks have the highest reported specific activity, primarily in the form of Cs-137. A preliminary assessment indicated external radiation dose rates for payload containers and/or for packaged payload assemblies from these tanks could potentially exceed applicable WIPP and CWC waste acceptance criteria. The scope of this study is limited to determining the feasibility of packaging the waste, such that external dose rates of payload containers and/or packaged payload assemblies meet CH-TRUM waste acceptance criteria as defined by DOE/WIPP-02-3122 and HNF-EP-0063. Based on the predicted range of densities and specific activities calculated in Appendix A, packaged waste

from the remaining seven SSTs meet acceptance criteria when processed and packaged in accordance with the baseline approach and were not considered in this analysis.

This study concludes that neither the current baseline process and container nor any of the three alternatives alone will be adequate to achieve a total packaged waste volume of 1.1 million gallons. The Cs-137 concentration (reported) for the upper waste layer of T-111 is the key factor contributing to excessive external waste package dose rates.

The results of Section 4.3.1 show that although it may be possible to package the upper layer of Tank T-111 waste in either SWBs or drums there is substantial risk that these containers will not meet external dose rate waste packaging and disposal requirements. In order to meet packaging requirements for 55-gallon drums, the upper layer of Tank T-111 waste must contain 70 to 75 wt% water with a minimum density of 1.12 g/ml. For packaging in SWBs, the upper layer of Tank T-111 waste must contain 80 to 85 wt% water with a minimum density of 1.07 g/ml. In either case, the upper layer of Tank T-111 waste will likely not pass the paint filter test unless sorbent is added. The measured water content of segment 1 from Tank T-111 core sample 33 was 80.35 wt%, with a density of 1.16 g/ml. No drainable liquid was collected in segment 1. However, it is likely that liquid could separate from Tank T-111 waste if subjected to centrifugation or vibration.

In addition homogenization of the upper layer of T-111 waste with the lower activity layers of T-111 to achieve a lower average Cs-137 concentration (T-111 average) entails significant risk and should not be considered a technically viable alternative. T-111 waste (even when assuming homogenous mixing) exceeds the CWC 100 mrem/hr dose rate limit when packaged in a SWB. Although it may be possible to package T-111 blended waste into 55-gallon drums, insufficient experience with the planned SST retrieval technologies exists to assume that homogenous mixing of T-111 can be accomplished. In addition waste slumping, where the upper layer slides to the bottom of the tank during retrieval operations, cannot be predicted or controlled. The occurrence of waste slumping poses a risk that external container dose rates, due to higher specific activity from T-111's top layer, will exceed WIPP or CWC external dose rate acceptance limits

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APPENDICES

Appendix A CH-TRUM Waste Shielding Analysis A-1

LIST OF TERMS

CH-TRUM	Contact Handled Transuranic Mixed
HLW	High Level Waste
PBI	Performance Based Initiative
PNNL	Pacific Northwest National Laboratory
RCRA	Resource Conservation and Recovery Act
SST	Single Shell Tanks
SWB	Standard Waste Boxes
TWINS	Tank Waste Information Netware System
WIPP	Waste Isolation Pilot Plant
WPS	Waste Packaging System
WTP	Waste Treatment Plant

1.0 Introduction

In FY 2004 CH2M HILL issued a multi-year contract to design and deploy a Contact Handled Transuranic Mixed (CH-TRUM) Waste Packaging System (WPS) to supplement the non-High Level Waste (HLW) treatment capacity provided by the Waste Treatment Plant (WTP). Procurement of this packaging system supports Performance Based Initiative (PBI-4). The goal is to package and prepare for disposition 1.1 million gallons of CH-TRUM waste by September 2006. The schedule of activities for design, deployment, start-up, and operation of the WPS takes place in fiscal years 2004-2006. Currently CH2M HILL is completing the Phase 3 final design review of the CH-TRUM WPS.

1.1 Background

Ten Single Shell Tanks (SST) have been identified as candidate CH-TRUM waste tanks. These tanks are: B-201, B-202, B-203, B-204, T-201, T-202, T-203, T-204, T-110, and T-111. Waste from these tanks will be retrieved, packaged, and ultimately disposed of at the Carlsbad Waste Isolation Pilot Plant (WIPP).

The packaging process uses dryer technology to dewater retrieved tank waste. The current technical baseline for the WPS specifies that the dewatered waste will be packaged in Standard Waste Boxes (SWB). However, over the past several months results of process development testing, vendor design reviews, and analyses conducted by the Pacific Northwest National Laboratory (PNNL) have indicated that modifications to the CH-TRUM WPS design and/or operational approach may be desirable to mitigate potential high radiation dose rate issues associated with packaged CH-TRUM waste.

1.2 Problem Definition

The WIPP acceptance criteria for CH-TRUM waste includes requirements that the external radiation dose rate of payload containers (e.g., SWB, 55-gallon drum) and packaged payload assemblies (i.e., TRUPACT-II) be less than or equal to 200 mrem/hour at contact, and that the external dose rate of payload assemblies be less than or equal to 10 mrem/hour at 2 meters (DOE/WIPP-02-3122). In addition, the Hanford Site Solid Waste Acceptance Criteria limits the external dose rate on payload containers to less than or equal to 100 mrem/hour at 30 centimeters (HNF-EP-0063).

The vacuum drying technology and dewatering process employed by the CH-TRUM WPS concentrates Cs-137 in the final dryer product and results in Cs-137 concentrations that are higher than those reported for the in-tank waste. In addition, an analysis of composite core samples obtained from Tank T-111 indicates that Cs-137 and Am-241 concentrations vary with the depth of waste in the tank. The analysis identified a surface layer (~ top 40 inches) with specific activity higher than the mean concentration reported in TWINS. Uncertainties associated with final packaged waste densities and increased specific activities as a consequence of the drying process, tank-specific contributions from Bremsstrahlung radiation, and increased

Cs-137 concentrations in T-111 may result in external dose rates from CH-TRUM payload containers and packaged payload assemblies that exceed the criteria introduced above. The high specific activity identified in the top layer of T-111, coupled with tank-specific contributions from Bremsstrahlung radiation confirmed the need for a more detailed dose rate evaluation of the waste from Tanks B-201, B-202, and T-111. Based on the high specific activity associated with the Tank T-111 waste, processing and packaging of the top layer of Tank T-111 (assuming blending does not occur during retrieval operations) may present the most significant challenge in terms of compliance with these criteria.

1.3 Purpose

The objective of this study is to identify technically feasible alternatives for dose rate mitigation that will allow the CH-TRUM WPS Project to package the maximum amount of CH-TRUM waste for eventual transportation and disposal at WIPP.

2.0 Scope

This study focuses on three tanks: B-201, B-202, and T-111. These tanks have the highest reported specific activity, primarily in the form of Cs-137. A preliminary assessment indicated external radiation dose rates for payload containers and/or for packaged payload assemblies from these tanks could potentially exceed applicable WIPP and CWC waste acceptance criteria. The scope of this study is limited to determining the feasibility of packaging the waste such that external dose rates of payload containers and/or packaged payload assemblies meet these waste acceptance criteria. This study addresses the current Waste Packaging System baseline process and packaging container (SWB) to determine the feasibility of treating and packaging 1.1 million gallons of CH-TRUM waste using the baseline approach. In addition, the study identifies three potential alternatives and evaluates the technical feasibility of these alternatives to mitigate potential high dose rate consequences resulting from the high Cs-137 concentrations in B-201, B-202, and T-111. The alternatives addressed by this study are:

- 55-gallon Drum Alternative. Evaluates the technical feasibility of packaging 1.1 million gallons of CH-TRUM waste using the WPS baseline process and 55-gallon drums as the payload containers. This alternative assumes a dryer product at 20 weight percent water and no sorbent addition.

- Moisture Content Alternative. Evaluates the technical feasibility of treating and packaging 1.1 million gallons of CH-TRUM waste by adjusting the moisture content of the WPS product to manage product densities and specific activities. This alternative assumes no sorbent addition to the dryer product. Both SWBs and 55-gallon drums are evaluated.

- Sorbents Addition Alternative. Evaluates the technical feasibility of treating and packaging 1.1 million gallons of CH-TRUM waste through process additive additions to manage WPS product densities and specific activities, for product packaged in both SWBs and 55-gallon drums, at assumed 20 weight percent water.

3.0 Basis of Assessment

3.1 Dose Rate Calculations

Appendix A provides the basis for the external radiation dose rate values used for this study and includes:

- Dose rate calculations and assumptions
- Bremsstrahlung determination
- MicroShield calculations

Dose rates were calculated at contact, 30 cm, and 2 m for a range of product densities (g/ml) and specific activities (µCi/ml) for various configurations of payload containers and packaged payload assemblies. External dose rates presented in the Appendix A tables have been corrected to account for Bremsstrahlung radiation and radionuclide decay (through July 2005).

3.2 Density Calculations

The estimated product densities for tanks B-201, B-202, and T-111 are extrapolated from laboratory testing of existing composite sludge samples. PNNL conducted laboratory scale vacuum drying of sludge composites from four tanks that are candidates for sludge retrieval and packaging using the CH-TRUM WPS (PNNL-14670, May 2004). The particle densities of the vacuum dried sludge composite samples were measured to estimate the product densities that may result from processing (i.e., vacuum drying) CH-TRUM waste. The resulting *particle density* data is presented in Table 3-1. The *particle density* is equivalent to the *bulk density* only if there is no void fraction present in the dried wastes. However, it is considered likely that the dried waste product will exhibit an appreciable void fraction due to the variety and irregularity of sludge particle geometries. In an effort to quantify the effect of void fraction on the bulk densities of the WPS product, CH2M HILL estimated the bulk densities of the product using the particle densities measured by PNNL staff and applying a void fraction value of approximately 0.38 (i.e., the void fraction at incipient fluidization in a fluidized bed)(CH2M HILL, March 2004). The results of the bulk density calculations, specified as Estimated Minimum Bulk Density, are also presented in Table 3-1.

Table 3-1. Particle and Estimated Minimum Bulk Densities of Vacuum Dried TRUM Tank Waste Composites (g/ml)

Tank	Particle Density (g/ml)	Est. Min. Bulk Density (g/ml)
T-110	2.36	1.46
B-203	2.34	1.45
T-203	2.39	1.48
T-204	2.64	1.64

Actual (post processing) packaged waste densities for the candidate CH-TRUM tanks have not been determined. The values presented in Table 3.1 represent a bounding range for the final product densities. Actual product bulk densities resulting from processing the sludge in the TRUM WPS may be greater than the estimated minimum bulk density (CH2M HILL, March 2004). However, the values tabulated above are considered a reasonable representation of the

range of bulk densities that may be achieved in the full scale TRUM WPS (i.e., approximately 1.4 g/ml to 2.7 g/ml). The estimated minimum bulk densities, calculated as discussed above, for the ten tanks which are candidates for sludge packaging by the TRUM WPS are provides in Appendix A, Table 3. The estimated product density (i.e., estimated minimum bulk density) for the three Tanks B-201, B-202, and T-111 are provided in Table 3-2.

Table 3-2. Tanks B-201, B-202, and T-111 Estimated Product Density and Specific Activity*

Source Term Values for 20% Moisture					
Tank	Estimated Product Density at 20 wt% water	Cs-137 Concentration (μCi/ml) at 20 wt% water	Bremsstrahlung Correction Factor	Decay Factor Corrected to June 1, 2005	Corrected for Decay and Bremsstrahlung (μCi/ml)
241-B-201	1.45 g/ml	0.48	1.20	0.90	0.52
241-B-202	1.45 g/ml	0.39	1.50	0.90	0.53
241-T-111 (upper layer)	1.46 g/ml	1.78	1.50	1.00	2.67
241-T-111 (average)	1.46 g/ml	0.51	1.50	0.90	0.69

*Values in this table are reproduced from Appendix A, Table 3.

Note: PNNL reported the vacuum dried tank waste samples were 100 wt% solids (PNNL-14670) For the purposes of this study, it is assumed that the moisture content of the dried product after packaging is 20 weight percent.

3.3 External Dose Rates, Product Density and Specific Activity

Several matrices (Appendix A) were developed to tabulate external dose rates for a range of specific activities and product densities for the ten candidate SSTs identified in Section 1.1. External dose rates were calculated for a combination of SWBs, 55 gallon drums, and TRUPACT-II containers at contact, 2 meters, and 30 centimeters. DOE/WIPP-02-3122 and HNF-EP-0063 establish waste package acceptance criteria that correspond to:

- Payload container external dose rate at or below 200 mrem/hr at contact
- Payload container external dose rate at or below 100 mrem/hr at 30 centimeters
- TRUPACT-II external dose rate at or below 200 mrem/hr at contact
- TRUPACT-II external dose rate at or below 10 mrem/hr at 2 meters

Six cases are represented in Appendix A that correspond to acceptable waste package payload container options and packaged payload assembly configurations:

1. Single 55 Gallon Drum (dose rate measured from the side)
2. TRUPACT-II container with a 7 drum pallet (dose rate measured from the bottom)
3. TRUPACT-II Container with a 7 drum pallet (dose rate measured from the side)
4. Single SWB (dose rate measured from the bottom)
5. TRUPACT-II Container with two SWBs (dose rate measured from the bottom)

6. TRUPACT-II Container with two SWBs (dose rate measured from the side)

The correlation of external dose rates to product densities and specific activities presented in Appendix A is the primary basis for evaluating dose rate mitigation alternatives. For clarity, Tables 3-3 and 3-4 summarize the applicable dose rates from Appendix A for the subject tanks and provide the interpolated dose rate values at the estimated minimum product bulk density and effective specific activity.

Table 3-3. SWB Dose Rate Summary

			Dose Rate (mrem/hr)**					
Tank	Est. Min Bulk Density	Effective Cs-137 Concentration @ Product 20 wt% H_2O *	Single SWB Bottom @ Contact	Single SWB Bottom @ 30 cm	2 SWBs in TRUPACT II Bottom @ Contact	2 SWBs in TRUPACT II Bottom @ 2 m	2 SWBs in TRUPACT II Side @ Contact	2 SWBs in TRUPACT II Side @ 2 m
	g/ml	µCi/ml	limit 200 mrem/hr	limit 100 mrem/hr	limit 200 mrem/hr	limit 10 mrem/hr	limit 200 mrem/hr	limit 10 mrem/hr
B-201	1.45	0.52	133	96	21	3	23	3
B-202	1.45	0.53	135	97	22	3	24	3
T-111 (average)	1.46	0.69	177	127	28	5	31	5
T-111 (upper layer)	1.46	2.67	>>200	>>100	142	21	183	24

* Corrected to include Bremsstrahlung and account for Decay to July 05

** Values for T-111 (upper layer) are linear extrapolations from Appendix A Table 3

Table 3-4. 55-Gallon Drum Dose Rate Summary

			Dose Rate (mrem/hr)**					
Tank	Est. Min Bulk Density	Effective Cs-137 Concentration @ Product 20 wt% H_2O*	Single Drum Side @ Contact	Single Drum Side @ 30 cm	7 Drums in TRUPACT II Bottom @ Contact	7 Drums in TRUPACT II Bottom @ 2 m	7 Drums in TRUPACT II Side @ Contact	7 Drums in TRUPACT II Side @ 2 m
	g/ml	µCi/ml	limit 200 mrem/hr	limit 100 mrem/hr	limit 200 mrem/hr	limit 10 mrem/hr	limit 200 mrem/hr	limit 10 mrem/hr
B-201	1.45	0.52	105	37	23	3	23	2
B-202	1.45	0.53	107	38	23	3	23	2
T-111 (average)	1.46	0.69	140	50	31	4	31	3
T-111 (upper layer)	1.46	2.67	>>200	182	157	19	157	14

* Corrected to include Bremsstrahlung and account for Decay to July 05

** Values for T-111 (upper layer) are linear extrapolations from Appendix A Table 3

4.0 Technical Assessment

This assessment evaluates the current WPS baseline process and packaging container (SWB) to determine the feasibility of treating and packaging 1.1 million gallons of CH-TRUM waste using the baseline approach. In addition, the study identifies three potential alternatives and evaluates the technical feasibility of these alternatives to mitigate potential high dose rate consequences resulting from the high Cs-137 concentrations in Tanks B-201, B-202, and T-111. For the purposes of this study, feasibility is defined in terms of four criteria:

Criterion 1 - Maintain payload container external dose rate ≤ 200 mrem/hr on contact

Criterion 2 - Maintain payload container external dose rate ≤ 100 mrem/hr at 30 centimeters

Criterion 3 - Maintain TRUPACT-II external dose rate ≤ 200 mrem/hr on contact

Criterion 4 - Maintain TRUPACT-II external dose rate ≤ 10 mrem/hr at 2 meters

The baseline process and container, or a dose mitigating alternative, is considered technically feasible if the packaged waste from B-201, B-202, and T-111 comply with the four criteria. These tanks have the highest Cs-137 concentration of the ten SSTs identified for CH-TRUM packaging. Based on the predicted range of densities and Cs-137 concentrations tabulated in Appendix A, the dryer product (packaged waste) from the remaining seven SSTs will meet all of the above criteria when processed and packaged in accordance with the baseline approach.

4.1 Baseline Process and Container (SWB)

The CH-TRUM WPS baseline process incorporates vacuum drying technology to dewater retrieved tank waste to approximately 20 weight percent water. The dewatered dryer product is packaged into SWBs. The addition of sorbents, and the associated effects on product densities and Cs-137 concentrations, is not considered for this baseline evaluation; sorbent addition is addressed in Section 4.4. Calculated dose rates for SWBs (across a range of product densities) are developed in Appendix A. Dose rates (interpolated values from Appendix A) for the estimated minimum bulk density and corrected specific activity are shown in Tables 4-1a, 4-1b, and 4-1c for the three SWB configurations presented in Appendix A:

- Configuration 1- Single SWB (as measured from the bottom) at contact and 30 cm
- Configuration 2- TRUPACT-II Container with 2 SWBs (as measured from the bottom) at contact and 2 meters
- Configuration 3- TRUPACT-II Container with 2 SWBs (as measured from the side) at contact and 2 meters

Table 4-1a. Single SWB (bottom dose rate)

Tank	Density (g/ml)	Specific Activity (µCi/ml)	Corrected Activity (µCi/ml)	Dose (mrem/hr)	
				Contact	30 cm
B-201	1.45	0.48	0.52	133	96
B-202	1.45	0.39	0.53	135	97
T-111 (top layer)	1.46	1.78	2.67	>>200	>>100
T-111 (averaged)	1.46	0.51	0.69	177	127

Table 4-1b. TRUPACT-II with Two SWBs (bottom dose rate)

Tank	Density (g/ml)	Specific Activity (µCi/ml)	Corrected Activity (µCi/ml)	Dose (mrem/hr)	
				Contact	2 m
B-201	1.45	0.48	0.52	21	3
B-202	1.45	0.39	0.53	22	3
T-111 (upper layer)	1.46	1.78	2.67	142	21
T-111 (average)	1.46	0.51	0.69	28	5

Table 4-1c. TRUPACT-II with Two SWBs (side dose rate)

Tank	Density (g/ml)	Specific Activity (µCi/ml)	Corrected Activity (µCi/ml)	Dose (mrem/hr)	
				Contact	2 m
B-201	1.45	0.48	0.52	23	3
B-202	1.45	0.39	0.53	24	3
T-111 (upper layer)	1.46	1.78	2.67	183	24
T-111 (average)	1.46	0.51	0.69	31	5

Tank B-201: The estimated product density of the packaged waste at 20 weight percent water is 1.45 g/ml. The corrected Cs-137 concentration at this density is 0.52 µCi/ml:

- Configuration 1 (Table 4-1a). Tank B-201 will meet Criteria 1 and 2 identified in Section 4.0 for CH-TRUM waste.

- Configuration 2 (Table 4-1b). Tank B-201 will meet Criteria 3 and 4 identified in Section 4.0 for CH-TRUM waste.

- Configuration 3 (Table 4-1c). Tank B-201 will meet Criteria 3 and 4 identified in Section 4.0 for CH-TRUM waste.

Note that dose rate values presented in the tables above, although conservative, are estimated values. At the estimated product values (1.45 g/ml and 0.52 µCi/ml) Tank B-201 will marginally pass Criterion 2 at 96 mrem/hr at 30 cm. Final product density will have a significant impact on

payload container external dose rates. Increased waste density will reduce the external dose rates; however, bulk densities less than or equal to 1.4 g/ml or Cs-137 concentrations greater than 0.6 µCi/ml will cause B-201 to fail Criterion 2 limits. Dispensation of the 100 mrem/hr at 30 cm limit at the CWC, although allowed by HNF-EP-0063, is not considered in this analysis.

Tank B-202: The assumed product density of the packaged waste at 20 weight percent water is 1.45 g/ml. The corrected Cs-137 concentration at this density is 0.53 µCi/ml:

- Configuration 1 (Table 4-1a). Tank B-202 will meet will meet Criteria 1 and 2 identified in Section 4.0 for CH-TRUM waste.

- Configuration 2 (Table 4-1b). Tank B-202 will meet will meet Criteria 3 and 4 identified in Section 4.0 for CH-TRUM waste.

- Configuration 3 (Table 4-1c). Tank B-202 will meet will meet Criteria 3 and 4 identified in Section 4.0 for CH-TRUM waste.

Based on the dose rate values presented in the table 4-1c (1.45 g/ml and 0.53 µCi/ml) Tank B-202 will marginally pass Criterion 2 (97 mrem/hr at 30 cm). If product densities are less than or equal to 1.4 g/ml or Cs-137 concentrations are greater than or equal to 0.6 µCi/ml B-202 will fail Criterion 2 limits. Dispensation of the 100 mrem/hr at 30 cm limit at the CWC, although allowed by HNF-EP-0063, is not considered in this analysis.

Tank T-111 (upper layer): The assumed product density of the packaged waste at 20 weight percent water is 1.46 g/ml. The corrected Cs-137 concentration at this density is 2.67 µCi/ml:

- Configuration 1 (Table 4-1a). Tank T-111 fails Criteria 1 and 2 identified in Section 4.0 for CH-TRUM waste.

- Configuration 2 (Table 4-1b). Tank T-111 fails Criterion 4 identified in Section 4.0 for CH-TRUM waste.

- Configuration 3 (Table 4-1c). Tank T-111 fails Criterion 4 identified in Section 4.0 for CH-TRUM waste.

Increased T-111 product densities will have no impact on dose rate mitigation. No consideration is made for blending of this layer with other tank layers (which exhibit notably lower specific activities) in T-111 during retrieval operations.

T-111 (average): The assumed product density of the packaged waste at 20 weight percent water is 1.46 g/ml. The corrected Cs-137 concentration, assuming homogeneous blending during retrieval, is 0.69 µCi/ml (CH2M HILL, April 2004):

- Configuration 1 (Table 4-1a). Tank T-111 blended tank waste meets Criterion 1 but fails Criterion 2 identified in Section 4.0 for CH-TRUM waste.

- Configuration 2 (Table 4-1b). Tank T-111 blended tank waste will meet Criteria 3 and 4 identified in Section 4.0 for CH-TRUM waste.

- Configuration 3 (Table 4-1c). Tank T-111 blended tank waste will meet Criteria 3 and 4 identified in Section 4.0 for CH-TRUM waste.

The results of the baseline process and container evaluation for the three tanks of interest are summarized in terms of the four feasibility criteria in Table 4-1d below.

Table 4-1d. Summary of Results for Baseline Process and Container

SWB (Baseline)	Criterion				Comments
	1	2	3	4	
B-201	P	P	P	P	Passes Criterion 2 with a 4% margin
B-202	P	P	P	P	Passes Criterion 2 with a 3% margin
T-111 (upper layer)	F	F	P	F	Cannot be packaged in a SWB
T-111 (average)	P	F	P	P	Requires blending of the tank waste to be packaged in SWBs

4.2 55-gallon Drum – Alternative 1

The 55-gallon Drum Alternative incorporates a payload container modification to the current CH-TRUM WPS baseline design. This alternative considers the baseline vacuum drying process that dewaters the waste to approximately 20 weight percent water, and packages the product in 55-gallon drums. The addition of sorbents, and the associated effects on product densities and Cs-137 concentrations, is not considered in this alternative; sorbent addition is addressed in Section 4.4. Calculated dose rates for 55-gallon drums (across a range of product densities) are developed in Appendix A. Dose rates (interpolated values from Appendix A) for the estimated minimum bulk density and corrected specific activity are shown in Tables 4-2a, 4-2b, and 4-2c for the three 55-gallon drum configurations presented in Appendix A:

- Configuration 1 - Single 55 Gallon Drum (as measured from the side) at contact and 30 cm
- Configuration 2 - TRUPACT-II container with a 7 drum pallet (as measured from the bottom) at contact and 2 meters
- Configuration 3 - TRUPACT-II Container with a 7 drum pallet (as measured from the side) at contact and 2 meters

Table 4-2a. Single 55-Gallon Drum (side dose rate)

Tank	Density (g/ml)	Specific Activity (µCi/ml)	Corrected Activity (µCi/ml)	Dose (mrem/hr)	
				Contact	30 cm
B-201	1.45	0.48	0.52	105	37
B-202	1.45	0.39	0.53	107	38
T-111 (top layer)	1.46	1.78	2.67	>>200	182
T-111 (average)	1.46	0.51	0.69	140	50

Table 4-2b. TRUPACT-II with Seven 55-Gallon Drums (bottom dose rate)

Tank	Density (g/ml)	Specific Activity (µCi/ml)	Corrected Activity (µCi/ml)	Dose (mrem/hr)	
				Contact	2 m
B-201	1.45	0.48	0.52	23	3
B-202	1.45	0.39	0.53	23	3
T-111 (top layer)	1.46	1.78	2.67	157	19
T-111 (average)	1.46	0.51	0.69	31	4

Table 4-2c. TRUPACT-II with Seven 55-Gallon Drums (side dose rate)

Tank	Density (g/ml)	Specific Activity (µCi/ml)	Corrected Activity (µCi/ml)	Dose (mrem/hr)	
				Contact	2 m
B-201	1.45	0.48	0.52	23	2
B-202	1.45	0.39	0.53	23	2
T-111 (top layer)	1.46	1.78	2.67	157	14
T-111 (average)	1.46	0.51	0.69	31	3

Tank B-201: The assumed product density of the packaged waste at 20 weight percent water is 1.45 g/ml. The corrected Cs-137 concentration at this density is 0.52 µCi/ml:

- Configuration 1 (Table 4-2a). Tank B-201 will meet Criteria 1 and 2 identified in Section 4.0 for CH-TRUM waste.
- Configuration 2 (Table 4-2b). Tank B-201 will meet Criteria 3 and 4 identified in Section 4.0 for CH-TRUM waste.
- Configuration 3 (Table 4-2c). Tank B-201 will meet Criteria 3 and 4 identified in Section 4.0 for CH-TRUM waste.

Tank B-202: The assumed product density of the packaged waste at 20 weight percent water is 1.45 g/ml. The corrected Cs-137 concentration at this density is 0.53 µCi/ml:

- Configuration 1 (Table 4-2a). Tank B-202 will meet Criteria 1 and 2 identified in Section 4.0 for CH-TRUM waste.
- Configuration 2 (Table 4-2b). Tank B-202 will meet Criteria 3 and 4 identified in Section 4.0 for CH-TRUM waste.
- Configuration 3 (Table 4-2c). Tank B-202 will meet Criteria 3 and 4 identified in Section 4.0 for CH-TRUM waste.

Tank T-111 (upper layer): The assumed product density of the packaged waste at 20 weight percent water is 1.46 g/ml. The corrected Cs-137 concentration at this density is 2.67 µCi/ml:

- Configuration 1 (Table 4-2a). Tank T-111 fails Criteria 1 and 2 identified in Section 4.0 for CH-TRUM waste.

- Configuration 2 (Table 4-2b). Tank T-111 fails Criterion 4 identified in Section 4.0 for CH-TRUM waste.

- Configuration 3 (Table 4-2c). Tank T-111 fails Criterion 4 identified in Section 4.0 for CH-TRUM waste.

Increased T-111 product densities will have no impact on dose rate mitigation. No consideration is made for blending of this layer with other tank layers (which exhibit notably lower specific activities) in T-111 during retrieval operations.

T-111 (average): The assumed product density of the packaged waste at 20 weight percent water is 1.46 g/ml. The corrected Cs-137 concentration, assuming homogeneous blending during retrieval, is 0.69 µCi/ml (CH2M HILL, April 2004):

- Configuration 1 (Table 4-2a). Tank T-111 blended tank waste will meet Criteria 1 and 2 identified in Section 4.0 for CH-TRUM waste.

- Configuration 2 (Table 4-2b). Tank T-111 blended tank waste will meet Criteria 3 and 4 identified in Section 4.0 for CH-TRUM waste.

- Configuration 3 (Table 4-2c). Tank T-111 blended tank waste will meet Criteria 3 and 4 identified in Section 4.0 for CH-TRUM waste.

The results of the baseline process and container evaluation for the three tanks of interest are summarized in terms of the four feasibility criteria in Table 4-3 below.

Table 4-2d. Summary of Results for Alternative 1

55 Gallon Drum	Criterion				Comments
	1	2	3	4	
B-201	P	P	P	P	Can be packaged in 55-gallon drums
B-202	P	P	P	P	Can be packaged in 55-gallon drums
T-111 (upper layer)	F	F	P	F	Cannot be packaged in 55- gallon drums
T-111 (average)	P	P	P	P	Requires blending of the tank waste to be packaged in 55-gallon drums

4.3 Moisture Content Variation- Alternative 2

Two scenarios were evaluated (as described below) to determine what effect the moisture content of the packaged waste product may have on payload container and packaged payload assembly dose rates, for Tanks B-201, B-202, and T-111.

4.3.1 Scenario MC1 – "Post-Saturation Addition"

The following scenario is based on the Estimated Product Density (Appendix A) for the three tanks at an assumed moisture content of 20 wt%. This analysis assumes:

- The product is *hydraulically saturated* at these conditions
- That additional moisture cannot be accommodated within the volume occupied by the product
- The density of the water is 1 g/ml
- The volume change as a result of mixing (delta Vm) is zero (for every volume of water added to the product there is a direct, corresponding increase in volume of the mixture)
- The specific activity of the water is 0
- The water does not have shielding effects

The MC1 scenario represents dilution as commonly understood and applied. For example, a 1:1 dilution (by volume) of product with water results in a mixture of twice the original product volume, a specific activity (i.e., µCi/ml) of ½ of the original product specific activity, and a decrease in density of the resulting mixture (assuming all possible product densities are greater than 1 g/ml). It should be noted however, that the actual waste may not perform as described above and this condition described is bounding. It should also be noted that the density is conservatively underestimated when compared to actual tank waste.

The impact of varying the moisture content of the WPS product, in accordance with the MC1 Scenario, for the three subject tanks is presented in Figures 4-1 through 4-4.

Figure 4-1. Tank B-201 Scenario MC1

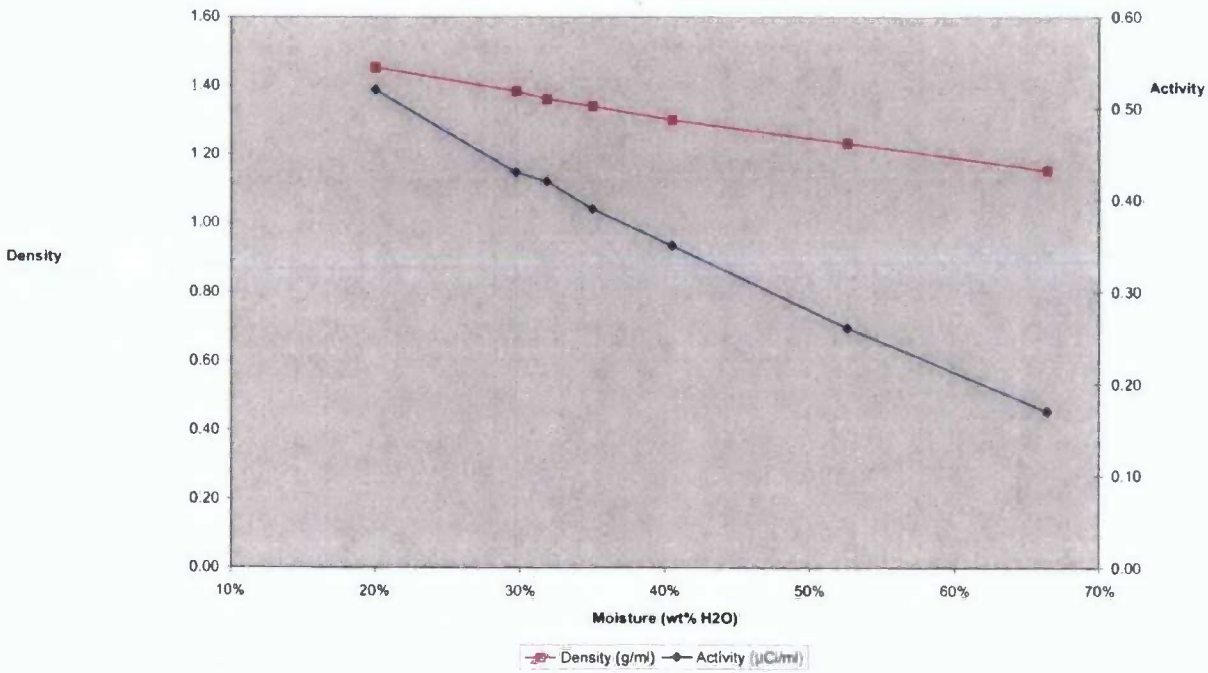


Figure 4-2. Tank B-202 Scenario MC1

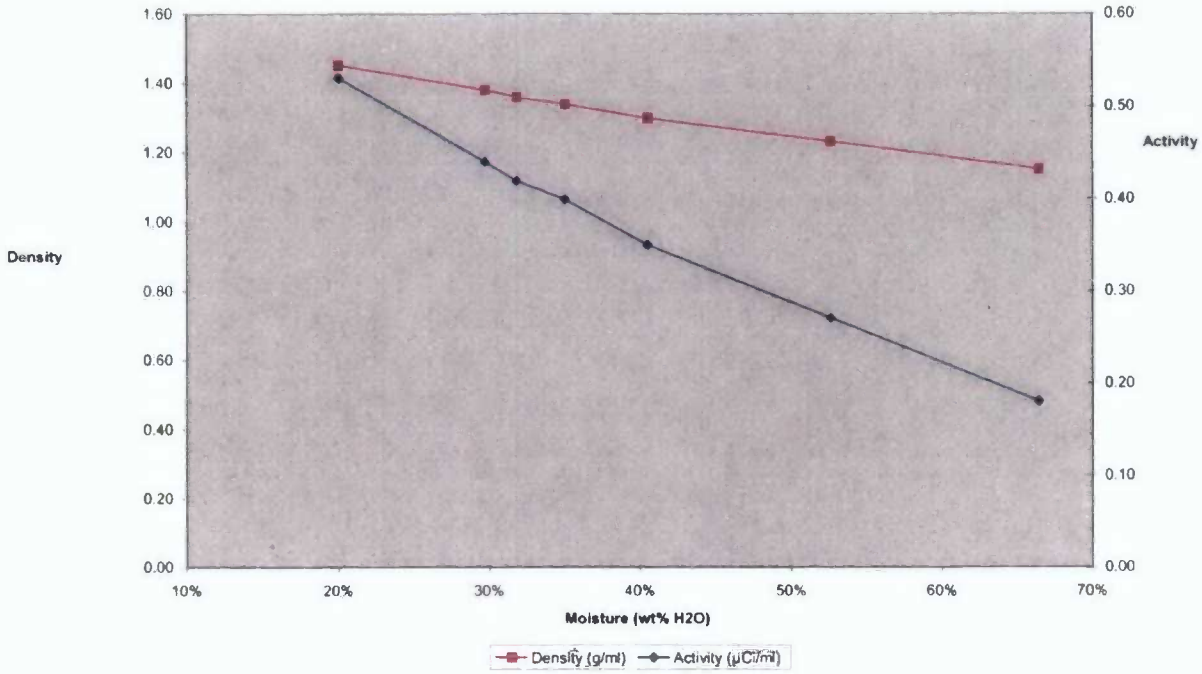


Figure 4-3. Tank T-111 (upper layer) Scenario MC1

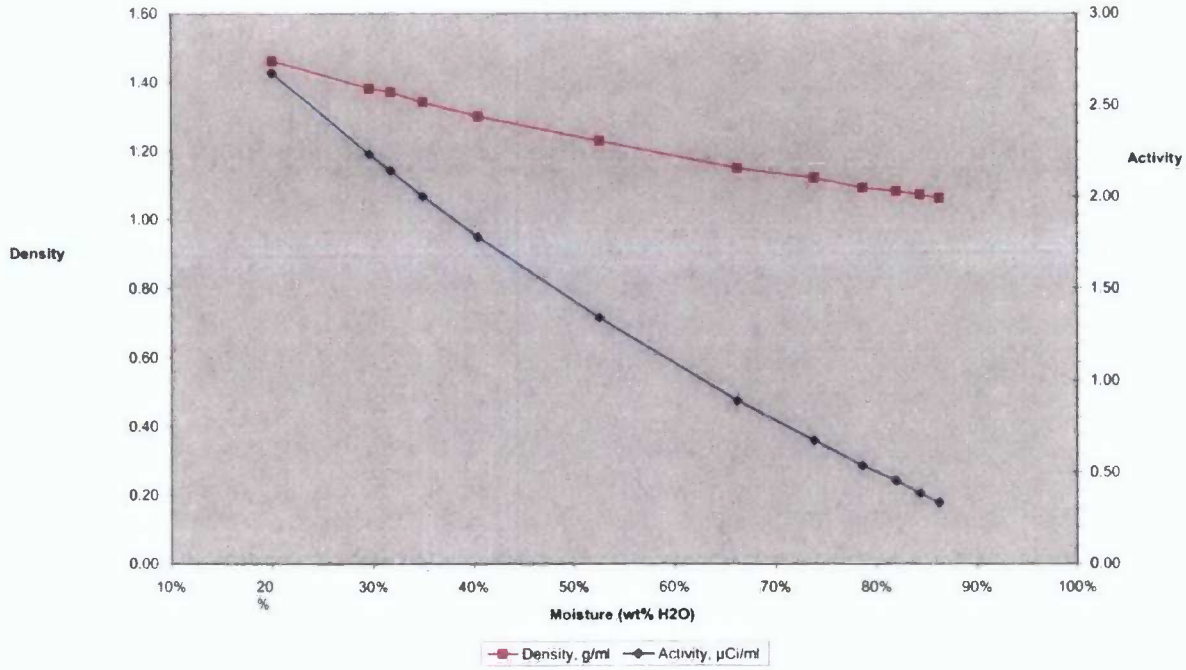
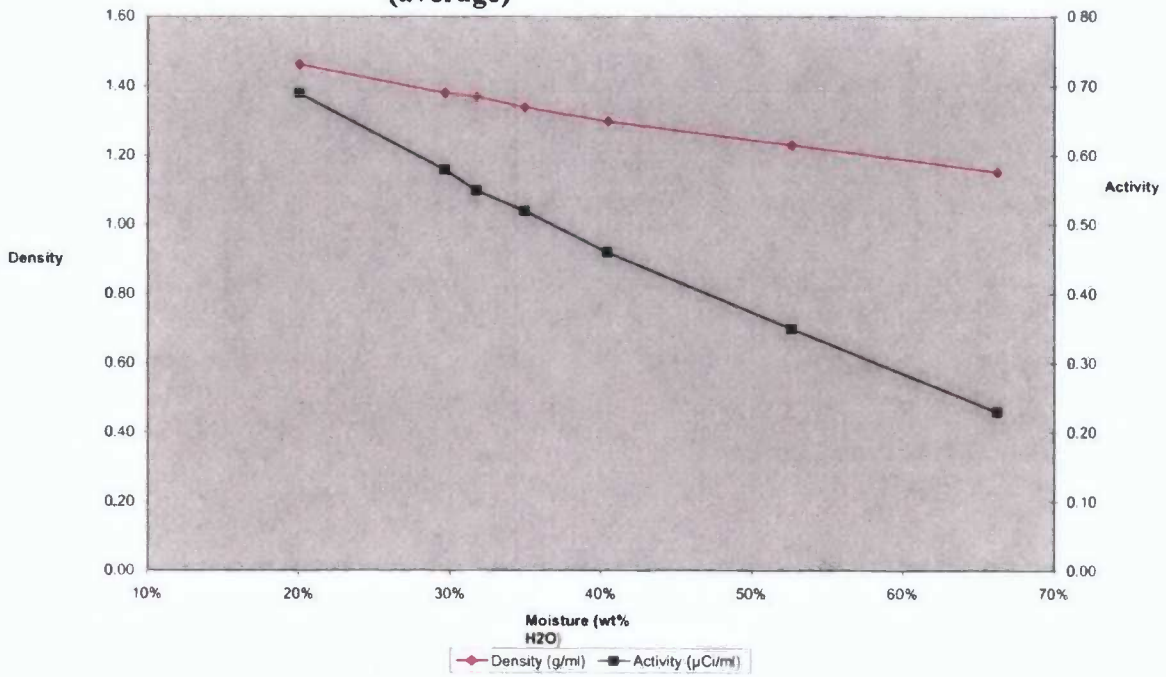


Figure 4-4. Tank T-111 (average)



While increasing moisture content in accordance with this scenario decreases the specific activity of the product and therefore dose rates associated with the waste containers and payload assemblies, it also decreases product density which has the opposite effect (i.e., increases container/payload assembly dose rates). However, the effect of decreasing the specific activity clearly predominates.

In accordance with the assumptions, approach and results of Scenario MC1, as correlated to applicable dose rate data in Appendix A, the following table identifies the approximate minimum moisture content range required to produce product that is compliant with all 4 criteria for the three tanks and two container types of interest.

Table 4-3. Scenario MC1- Minimum Moisture Content (wt% water) Required to Meet Dose Rate Criteria

Tank	Container	Minimum Moisture Content Range	Associated Density (approx)
B-201	SWB Drum	NR (20 wt%) NR (20 wt%)	1.45 g/ml 1.45 g/ml
B-202	SWB Drum	NR (20 wt%) NR (20 wt%)	1.46 g/ml 1.45 g/ml
T-111 (upper layer)	SWB Drum	80-85 wt% 70-75 wt%	1.07 g/ml 1.12 g/ml
T-111 (average)	SWB Drum	35-40 wt% NR	1.32 1.46

NR – Moisture content optimization not required; product may be successfully packaged at design basis conditions (i.e., 20 wt% water).

4.3.2 Scenario MC2 – "Pre-Saturation Addition"

Similar to the MC1, this scenario uses the "Estimated Product Density" for the three subject tanks and an assumed (baseline) moisture content of 20 wt% water. However, this alternative assumes that the product is *hydraulically unsaturated* at these baseline conditions.

Accordingly, this scenario assumes that the void fraction used in the calculation of estimated minimum bulk densities, i.e., approximately 0.38, is available to accommodate water, *without a corresponding increase in the volume of the mixture*. That is, for every 100 ml of product, 38 ml is void space and available to accommodate 38 ml of water with no increase in volume of the resulting product. Under this scenario, increasing moisture content increases the density of the product. It should be noted however, that the actual waste may not perform as described above and this condition described is bounding. It should also be noted that the density is conservatively underestimated when compared to actual tank waste. The results associated with this scenario are presented in Table 4-4.

Table 4-4. Scenario MC2- Final Moisture Content (wt% water) and Associated Density

Tank	Initial % Moisture	Initial Density	Final % Moisture	Final Density
B-201	20	1.45 g/ml	36.6	1.83 g/ml
B-202	20	1.45 g/ml	36.6	1.83 g/ml
T-111 (upper layer)	20	1.46 g/ml	36.5	1.84 g/ml
T-111 (average)	20	1.46 g/ml	36.5	1.84 g/ml

Under this scenario, the additional water is assumed to have no shielding effects. Accordingly, the specific activity of the mixture, or final product, is identical to that of the original product. This scenario has no effect on the specific activity of the product, but serves to increase final product density substantially. As previously noted, increasing product density results in decreasing dose rate associated with the payload containers and payload assemblies.

In accordance with the assumptions, approach and results of Scenario MC2, as correlated to applicable dose rate data in Appendix A, the following table identifies the degree of compliance with the 4 criteria for the three tanks and two container types of interest.

Table 4-5. Results of Scenario MC2

Tank	Container	Pass (all Criteria)	Fail	Criteria Failed
B-201	SWB Drum	X X		
B-202	SWB Drum	X X		
T-111 (upper layer)	SWB Drum		X X	1, 2, 4, Note 1 1, 2, 4, Note 1
T-111 (average)	SWB Drum	 X	X	2

Note 1 – Available dose rate data does not enable conclusive evaluation of compliance with Criterion 3 at specific activities exceeding 1.6 μCi/ml

4.4 Sorbent Addition – Alternative 3

To assess the potential impacts that the addition of sorbents to the packaged waste product may have on payload container and payload assembly dose rates, three scenarios were evaluated (as described below) for the three tanks which may present high dose rate concerns. Note that the dose rate calculations associated with the three sorbent scenarios assumes a 20 wt% moisture content in the resulting waste products (including the sorbents).

4.4.1 Scenario S1 – 50 vol% Low Density Sorbent Addition

This scenario involves the use of a low-density sorbent in a 1:1 mix (by volume) with the waste product. The assumed sorbent density is 0.7 g/ml. The effect of this sorbent addition is to decrease product density and decrease product specific activity. The anticipated results of applying this scenario to wastes from Tanks B-201, B-202, and T-111 (upper layer) are presented in Table 4-6. Note that the specific activities presented in this table have been adjusted with applicable tank-specific Bremsstrahlung and decay correction factors (Appendix A).

Table 4-6. Scenario S1 Low Density Sorbent Addition

50% (by volume) 0.7 gm/ml sorbent		
Density	Rad µCi/ml)	Tank
1.1	0.26	B-201
1.1	0.27	B-202
1.1	1.34	T-111 (upper layer)
1.1	0.35	T-111 (average)

The results of this scenario for the three tanks of interest and two container types are summarized in terms of the four feasibility criteria in Table 4-7 below.

Table 4-7. Results of Scenario S1

Tank	Container	Pass	Fail	Criteria Failed
B-201	SWB Drum	X X		
B-202	SWB Drum	X X		
T-111 (upper layer)	SWB Drum		X X	1, 2, 4 1, 2
T-111 (average)	SWB Drum	X X		

4.4.2 Scenario S2 – 25% Sorbent/25% Sand Addition

This scenario involves the use of a low-density sorbent and sand as additives to the waste. The resulting product is assumed to be 25 vol% sorbent, 25 vol% sand, and 50 vol% waste. The density of the sorbent is assumed to be 0.7 g/ml; the density of the sand is assumed to be 1.8 g/ml. Because the "effective dilution" associated with this scenario is identical to Scenario S1 (i.e., an overall 1:1 mixture of waste with sorbent/sand additive), its effect, in terms of specific activity reduction, is identical to Scenario S1 as well. However, the inclusion of sand (which has a relatively high density) results in a product with a higher density relative to Scenario S1. As previously noted, a higher density is advantageous in terms of dose rate reduction due to the self shielding effect. The anticipated results of applying this scenario to the waste from the three Tanks B-201, B-202, and T-111 (upper layer) are presented in Table 4-8. Note that the specific activities presented in this table have been adjusted with applicable tank-specific Bremsstrahlung and decay correction factors (Appendix A).

Table 4-8. Scenario S2 Sorbent and Sand Addition

25% Sorbent, 25% 1.8 g/ml sand, 50% product		
Density	Rad (μCi/ml)	Tank
1.3	0.26	B-201
1.2	0.27	B-202
1.2	1.34	T-111 (upper layer)
1.2	0.35	T-111 (average)

The results of this scenario for the three tanks of interest and two container types are summarized in terms of the four feasibility criteria in Table 4-9 below.

Table 4-9. Results of Scenario S2

Tank	Container	Pass	Fail	Criteria Failed
B-201	SWB Drum	X X		
B-202	SWB Drum	X X		
T-111 (upper layer)	SWB Drum		X X	1, 2, 4 1, 2
T-111 (average)	SWB Drum	X X		

4.4.3 Scenario S3 - 20% Sorbent/40% Sand Addition

This scenario involves the use of the low-density sorbent and sand described above, but in different proportions. The resulting product is assumed to be 20 vol% sorbent, 40 vol% sand, and

40 vol% waste. This scenario results in a higher "effective dilution" of the waste relative to Scenarios S1 and S2 because the resulting product is (only) 40 vol% waste. Additionally, due to the relatively high sand content, this scenario exhibits higher product densities than the other two scenarios. The anticipated results of applying this scenario to the waste from Tanks B-201, B-202, and T-111 (upper layer) are provided in Table 4-10. Note that the specific activities presented in this table have been adjusted with applicable tank-specific Bremsstrahlung and decay correction factors (Appendix A).

Table 4-10. Scenario S3 Sorbent and Sand Addition

20% Sorbent, 40%(1.8 g/ml) Sand, 40% product		
Density	Rad (μCi/ml)	Tank
1.4	0.20	B-201
1.4	0.21	B-202
1.4	1.07	T-111 (upper layer)
1.4	0.28	T-111 (average)

The results of this scenario for the three tanks of interest and two container types are summarized in terms of the four feasibility criteria in Table 4-11 below.

Table 4-11. Results of Scenario S3

Tank	Container	Pass	Fail	Criteria Failed
B-201	SWB Drum	X X		
B-202	SWB Drum	X X		
T-111 (upper layer)	SWB Drum		X X	1, 2 1
T-111 (average)	SWB Drum	X X		

5.0 Conclusions

This evaluation focused on three SSTs; B-201, B-202, and T-111. The volume of waste represented by these tanks is included in the 1.1 million gallons of waste to be packaged by the CH-TRUM Waste Packaging Project. A preliminary assessment, due to high specific activity, indicated external radiation dose rates for payload containers and/or for packaged payload assemblies could potentially exceed applicable WIPP and CWC waste acceptance criteria. The WIPP acceptance criteria for CH-TRUM waste includes requirements that the external radiation dose rate of payload containers (e.g., SWB, 55-gallon drum) and payload assemblies (i.e., TRUPACT-II) be less than or equal to 200 mrem/hour at contact, and that the external dose rate of payload assemblies be less than or equal to 10 mrem/hour at 2 meters (DOE/WIPP-02-3122). In addition, the Hanford Site Solid Waste Acceptance Criteria limits the external dose rate on payload containers to less than or equal to 100 mrem/hour at 30 centimeters (HNF-EP-0063).

The WPS baseline process and three potential dose rate mitigation alternatives were evaluated to determine whether high dose rate issues associated with Tanks B-201, B-202, and T-111 could be addressed. These alternatives were:

- Alternative 1 - Dryer product packaged in 55-gallon drums
- Alternative 2 - Varying the moisture content of the dryer product packaged in SWBs and 55-gallon drums
- Alternative 3 - Addition of sorbent to the dryer product packaged in SWBs and 55-gallon drums

The objective was to investigate technically feasible packaging and/or process alternatives that would allow the CH-TRUM WPS Project to package the maximum amount of CH-TRUM waste for eventual transportation and disposal at WIPP.

For the purposes of this study, four criteria were established to evaluate technical feasibility:

- Criterion 1 - Maintain payload container external dose rate $\leq$ 200 mrem/hr on contact
- Criterion 2 - Maintain payload container external dose rate $\leq$ 100 mrem/hr at 30 centimeters
- Criterion 3 - Maintain TRUPACT-II external dose rate $\leq$ 200 mrem/hr on contact
- Criterion 4 - Maintain TRUPACT-II external dose rate $\leq$ 10 mrem/hr at 2 meters

The baseline process and container and/or an alternative was considered technically feasible if the CH-TRUM WPS Project could treat and package, in compliance with the above four criteria, 1.1 million gallons of CH-TRUM waste.

Baseline Process and Container (SWB): The baseline process employs a "dry batch" technique whereby a bed substrate material (e.g. vermiculite) is first introduced into the dryer. Waste slurry

is then periodically metered into the drying process such that the bulk mixture liquid content remains less than or equal to 20 wt% water. The baseline process packaging container is an SWB. Extraction of solids from the dryer for packaging never empties the dryer, always leaving a bed of dry solids to initiate subsequent slurry addition and drying. Over time the drying process requires little or no addition of additional substrate material. Consequently, the use of additives and the associated effects on product density and specific activity are not considered in the baseline process.

External dose rates were evaluated, as measured from a single SWB and from a TRUPACT-II (containing two SWBs), with respect to the WIPP acceptance criteria and Hanford Site Solid Waste acceptance criteria for CWC. Based on expected product density and specific activity, only Tanks B-201 and B-202 meet all the criteria established in Section 4.0. Tank T-111 will not meet WIPP and CWC external dose rate requirements when packaged into SWBs:

- T-111 (upper layer) SWB exceeds WIPP criteria 200 mrem/hr at contact
- T-111 (upper layer) SWB exceeds CWC criteria 100 mrem/hr at 30 cm
- T-111 (upper layer) TRUPACT-II exceeds WIPP criteria 10 mrem/hr at 2 m
- T-111 (average) SWB exceeds CWC Criteria 100 mrem/hr at 30 cm

Tank T-111 represents approximately 447,000 gallons of CH-TRUM waste. ***The CH-TRUM Waste Packaging Project cannot successfully package 1.1 million gallons of tank waste using the baseline process in SWBs for eventual disposition, shipment and disposal at the WIPP.***

It should be noted that Tanks B-201 and B-202 marginally pass CWC external payload dose rate limits (Criterion 2). Based on the uncertainty associated with the final packaged waste densities and specific activity there is a risk that waste from these tanks would not be accepted at the CWC. Although dispensation of the 100 mrem/hr at 30 cm limit at the CWC is allowed by HNF-EP-0063, it is not considered as an alternative for this analysis.

Alternative 1: Alternative 1 evaluates mitigating potential high dose rate issues by packaging CH-TRUM waste in 55-gallon drums. This alternative assumes the same dewatering process as Alternative 1 with dryer product being packaged at 20 weight percent moisture. The use of sorbents and the associated effects on product density and specific activity was not considered in this alternative. Alternative 1 evaluated the external dose rate, as measured from a single 55-gallon drum and from a TRUPACT-II containing seven 55-gallon drums, with respect to the WIPP acceptance criteria and Hanford site solid waste acceptance criteria for CWC.

Based on expected product density and specific activity, Tanks B-201, B-202, and Tank T-111 (average) meet all of the criteria established in Section 4.0.

Waste packaged from the upper layer of Tank T-111 will not meet WIPP and CWC external dose rate requirements when packaged in 55-gallon drums:

- T-111 (upper layer) single 55-gallon drum exceeds WIPP criteria 200 mrem/hr at contact
- T-111 (upper layer) single 55-gallon drum exceeds CWC criteria 100 mrem/hr at 30 cm
- T-111 (upper layer) TRUPACT-II with seven 55-gallon drums exceeds WIPP criteria 10 mrem/hr at 2 m

The Tank T-111 (average) reported specific activity assumes blending of the tank waste (homogenous mixing of the upper waste layer with the lower activity layers) to achieve a lower mean specific activity. Reliance on homogenous blending during retrieval carries significant risk. SST retrieval technologies have not been demonstrated or proven in the field and there is no basis (retrieval experience) to assume retrieval operations can be adapted to homogenize T-111 to achieve specific activities that are representative of the reported tank average. In addition, phenomenon such as slumping, where the upper layer slides to the bottom of the tank during retrieval operations cannot be predicted or controlled. Should slumping of a significant portion of the top layer occur there is a significant risk that the packaging system would receive waste with a specific activity that would cause external dose rates to exceed acceptance limits.

Tank T-111 represents approximately 447,000 gallons of CH-TRUM waste. ***The CH-TRUM Waste Packaging Project cannot successfully package 1.1 million gallons of tank waste in 55-gallon drums using the current baseline process for eventual disposition, shipment and disposal at the WIPP.***

Alternative 2: The potential impact of increasing the moisture content was evaluated using two different scenarios. The first, scenario MC1 (post-saturation), indicates that increasing moisture content has the effect of decreasing waste specific activity and therefore container/packaged payload assembly dose rate, and decreasing product density. Both effects are corollary to the direct volume increase associated with water addition under this alternative. However the reduction of specific activity clearly predominates.

Based on the results of the MC1 analysis, wastes from Tanks B-201 and B-202 may be compliantly packaged at baseline conditions (i.e., moisture content optimization is not required). Waste from Tank T-111 may meet WIPP and CWC external dose rate requirements if product moisture content is optimized as follows:

- T-111 (upper layer) requires moisture content optimization (80-85 wt%) to meet Section 4.0 criteria when packaged in SWBs
- T-111 (upper layer) requires moisture content optimization (70-75 wt%) to meet Section 4.0 criteria when packaged in 55-gallon drums
- T-111 (average) requires moisture content optimization (35-40 wt%) to meet Section 4.0 criteria when packaged in SWBs

The second scenario, MC2 (pre-saturation), is in effect; the direct opposite of MC1 in that additional water does not increase product volume nor alter its specific activity. The additional

moisture is accommodated by the void fraction in the "dry" product. The effect of increasing moisture content of the TRUM WPS waste product under this scenario is solely an increase in product density. Note however, that increasing product density results in decreasing the dose rate of the TRUM waste containers and packages. Based on the results of the MC2 analysis, Tanks B-201 and B-202 meet all Section 4.0 criteria at baseline conditions when packaged in SWBs and 55-gallon drums (i.e., moisture content optimization is not required). Under the MC2 scenario, waste from Tank T-111 will not meet WIPP and CWC external dose rate requirements:

- T-111 (upper layer) single SWB exceeds WIPP criterion 200 mrem/hr at contact
- T-111 (upper layer) single SWB exceeds CWC criterion 100 mrem/hr at 30 cm
- T-111 (upper layer) SWB/TRUPACT-II exceeds WIPP criterion 10 mrem/hr at 2 m
- T-111 (upper layer) single 55-gallon drum exceeds WIPP criterion 200 mrem/hr at contact
- T-111 (upper layer) single 55-gallon drum exceeds CWC criterion 100 mrem/hr at 30 cm
- T-111 (upper layer) Drum/TRUPACT-II exceeds WIPP criterion 10 mrem/hr at 2 m
- T-111 (average) single SWB exceeds CWC criterion 100 mrem/hr at 30 cm

Qualitatively, the results indicate that increasing the moisture content of the TRUM WPS waste product may be an effective method to mitigate potential high dose rate issues associated with processing and packaging TRUM sludge retrieved from Tank T-111. Note however, that the moisture content required to successfully package waste from T-111 using SWB, in accordance with Scenario MC1, is approximately 80-85 wt% water. This moisture content level is associated with a final product that is one part waste (at 20 wt% water) and approximately 6 parts water (by volume) in accordance with the MC1 model. Similarly, should packaging in a drum be desirable, the required product moisture content of 70-75 wt% represents a final product that is one part waste (at 20 wt% water) and approximately 3 parts water (by volume).

Tank T-111 in-tank waste moisture content is documented in TWINS at approximately 75 wt%. This tank has been interim stabilized. Accordingly, one may consider that all free liquid has been removed as an approximation. The remaining "solid" waste (i.e., no free liquid) exhibits the aforementioned moisture content (i.e., approximately 75 wt%). Therefore, to package this waste in a SWB may require that very little water be removed from the waste slurry as retrieved, and that sorbents be added to mitigate free liquid concerns. While it is possible that mitigation of high dose rate issues associated with this tank may involve a moisture content optimization element, it is not probable that such optimization solely will result their effective mitigation.

In addition, products with high moisture contents involve "free liquid" concerns. To address this concern, tank waste composites from tanks B-203, T-110, T-203 and T-204, at a variety of dilutions, were subjected to modified paint filter liquids testing (SW-846, Method 9095A)(EPA 1994) to determine whether free liquid existed in any of the samples. The testing, conducted by PNNL, indicated that "30-wt% solids" dilutions of composite waste samples from Tanks T-203 and T-204 passed the paint filter test; corresponding composite waste samples from Tanks T-110 and B-204 did not pass the paint filter test (PNNL-14365). Nonetheless, this information suggests that products exhibiting moisture contents of up to approximately 70 wt% may not exhibit free liquids.

In accordance with the results of the MC1 Scenario, waste from Tanks B-201and B-202 may be compliantly packaged with the baseline process and container; no moisture content optimization is necessary. Tank T-111 (average) waste can be compliantly packaged in 55 gallon drums without moisture optimization. Results of the MC1 Scenario indicate that wastes from the upper layer of Tank T-111 may be successfully packaged if product moisture content is optimized in the range of 80-85 wt% for SWB or 70-75 wt% for drums. However, it is unlikely that waste from this tank could be packaged in SWBs or 55-gallon drums by optimizing moisture content solely; substantial sorbent addition may be required as well. For the Tank T-111 (average) case, a moisture content of 35-40 wt% would be required for packaging in a SWB.

Tank T-111 represents approximately 447,000 gallons of CH-TRUM waste. ***It is not probable that the CH-TRUM Waste Packaging Project can package 1.1 million gallons of tank waste for eventual disposition, shipment and disposal at the WIPP if alternate dewatering technology alone is employed to control the moisture content of Tank T-111.***

Alternative 3: To assess the potential impacts that the addition of sorbents to the packaged waste product may have on payload container and packaged payload assembly external dose rates, three scenarios were evaluated: S1 (50 volume percent low density sorbent addition), S2 (25 volume percent sorbent plus 25 volume percent sand addition), and S3 (20 volume percent sorbent and 40 volume percent sand addition). This Alternative assumed 20 wt% moisture content in the resulting waste products (including the additives). Based on the results of this analysis Tanks B-201, B-202 and the Tank T-111 (average) case may be packaged successfully in accordance with any of the three scenarios considered, using either SWBs or 55-gallon drums.

Waste from Tank T-111 (upper layer), when packaged in a SWB, fails at least two of the four criteria in all three scenarios considered (i.e., Criteria 1 and 2). In addition, the waste from Tank T-111 (upper layer) when packaged in 55-gallon drums, fails at least one of the four criteria in all three scenarios considered (i.e., Criterion 1):

<u>S1 (50 volume percent low density sorbent addition)</u>

- T-111 (upper layer) single SWB exceeds WIPP criterion 200 mrem/hr at contact
- T-111 (upper layer) single SWB exceeds CWC criterion 100 mrem/hr at 30 cm
- T-111 (upper layer) SWB/TRUPACT-II exceeds WIPP criterion 10 mrem/hr at 2 m
- T-111 (upper layer) single 55-gallon drum exceeds WIPP criterion 200 mrem/hr at contact
- T-111 (upper layer) single 55-gallon drum exceeds CWC criterion 100 mrem/hr at 30 cm

<u>S2 (25 volume percent sorbent plus 25 volume percent sand addition)</u>

- T-111 (upper layer) single SWB exceeds WIPP criterion 200 mrem/hr at contact
- T-111 (upper layer) single SWB exceeds CWC criterion 100 mrem/hr at 30 cm
- T-111 (upper layer) SWB/TRUPACT-II exceeds WIPP criterion 10 mrem/hr at 2 m
- T-111 (upper layer) single 55-gallon drum exceeds WIPP criterion 200 mrem/hr at contact
- T-111 (upper layer) single 55-gallon drum exceeds CWC criterion 100 mrem/hr at 30 cm

S3 (20 volume percent sorbent and 40 volume percent sand addition)

- T-111 (upper layer) single SWB exceeds WIPP criteria 200 mrem/hr at contact
- T-111 (upper layer) single SWB exceeds CWC criteria 100 mrem/hr at 30 cm
- T-111 (upper layer) single 55-gallon drum exceeds WIPP criterion 200 mrem/hr at contact

Succinctly, waste from the upper layer of Tank T-111 may not be compliantly packaged under any of the sorbent addition scenarios considered, regardless of payload container type. It may be possible to increase or modify the ratios of sorbent and sand to facilitate compliance, however such optimization is beyond the scope of this evaluation. Furthermore, the addition of additives (e.g., sorbents and fillers) entails potential regulatory, operations and cost considerations. If additives are introduced to address the "no" free liquids requirements of the WIPP WAC, or facilitate effective processing of the waste (as in the case of sorbents or fillers used as dry bed substrate in the drying process), then additive addition is compatible with applicable regulatory requirements. However, additives cannot be used solely to accomplish dilution; such dilution is prohibited under the WIPP WAC as well as RCRA.

The use of additives may result in an increased product volume which may, but does not necessarily result in the need for an increased number of waste containers and waste shipments. The use of additives may also adversely impact process throughput as a result of decreased processing rates and/or the incremental increase in process complexity and associated operational requirements.

Tank T-111 represents approximately 447,000 gallons of CH-TRUM waste. ***It is not probable that the CH-TRUM Waste Packaging Project can package 1.1 million gallons of tank waste for eventual disposition, shipment and disposal at the WIPP if T-111 is packaged using a modified process introducing additives in accordance with Scenarios 1 through 3.***

Based on the results of this study, neither the baseline process nor any of the alternatives alone will be adequate to achieve a total packaged waste volume of 1.1 million gallons. The Cs-137 concentration (reported) for the upper waste layer of T-111 is the key factor contributing to excessive external waste package dose rates. Reliance on blending of the upper layer of T-111 waste with the lower activity layers of T-111 to achieve a lower average Cs-137 concentration (T-111 average) entails significant project risk. Sufficient experience with the planned SST retrieval technologies does not exist to assume that homogenous mixing of T-111 can be accomplished. In addition, waste slumping, where the upper layer slides to the bottom of the tank during retrieval operations, cannot be predicted or controlled. The occurrence of waste slumping poses a risk that external container dose rates, due to higher specific activity from T-111's top layer, will exceed WIPP or CWC external dose rate acceptance limits. T-111 waste, even when assuming homogonous mixing, will exceed the CWC 100 mrem/hr dose rate limit when packaged in a SWB.

Tanks B-201 and B-202 marginally pass CWC external payload dose rate limits when packaged in SWBs. As a minimum, due to the uncertainties associated with the final packaged waste densities and specific activities for these two tanks, the project should consider packaging the CH-

TRUM waste into 55-gallon drums. Although switching waste containers alone will not allow the project to package T-111 upper layer waste, external container dose rates are significantly reduced when the waste is packaged in 55-gallon drums.

This study focused on determining the technical feasibility to package CH-TRUM waste such that payload containers and/or packaged payload assemblies would meet WIPP and CWC external dose rate acceptance criteria. The objective was to identify a technically feasible alternative(s) to allow the CH-TRUM WPS Project the flexibility to package the maximum amount of CH-TRUM waste for eventual transportation and disposal at WIPP. Implementation and impact to life-cycle project costs were not considered in the scope of this analysis. These include:

- The impact to TRU limits (100 nCi/g) with the addition of sorbents and sand
- Replace T-111 with alternative tank(s)
- Blending T-111 waste in WPS receipt tanks with lower specific activity waste from any of the other CH-TRUM waste tanks
- Cost associated with changing the waste package container
  - design
  - container cost
  - material handling
  - sampling
  - shipping
  - WIPP disposal
- Optimization of process parameters and/or a combination of alternatives (e.g., moisture content optimization with sorbent additive) to reduce high dose rates resulting from T-111 upper layer waste
- Cost associated with optimization and/or modifying process parameters to package T-111 upper layer waste

It may be possible to modify the ratios of sorbent and sand to reduce external radiation dose rates produced by T-111 upper layer waste. However, the use of additives entails potential regulatory considerations. If additives are introduced to address the "no" free liquids requirements of the WIPP WAC, or to facilitate effective processing of the waste then additive addition may be compatible with applicable regulatory requirements. Additives however cannot be used solely to accomplish dilution; such dilution is prohibited under the WIPP WAC as well as RCRA.

6.0 References

1. CH2M HILL, Aril 2004, *Change in Cesium-137 Concentrations Based on Water Content for Transuranic Tank Wastes*, CH2M HILL Hanford Group interoffice memorandum dated April 29, 2004 from Closure Projects Engineering Support to A. Tedeschi, CH2M HILL Hanford Group, Richland, Washington

2. CH2M HILL, March 2004, *Density Estimate for Packaged Transuranic Tank Waste*, CH2M HILL Hanford Group interoffice memorandum dated March 23, 2004 from Closure Projects Engineering Support to A. Tedeschi, CH2M HILL Hanford Group, Richland, Washington

3. DOE/WIPP-02-3122, Rev. 1, *Contact-Handled Transuranic Waste Acceptance Criteria for the Waste Isolation Pilot Plant*

4. HNF-EP-0063, Revision 10, *Hanford Site Solid Waste Acceptance Criteria*

5. PNNL, March 2004, *Density Measurements*, Battelle memorandum dated March 18, 2004 from J. Tingey to L. Bagaasen, Battelle Pacific Northwest Division, Richland, Washington

6. PNNL-14365, *Physical Property and Rheological Testing of Actual Transuranic Waste from Hanford Single-Shell Tanks*, J. Tingey et. al., August 2003, Pacific Northwest National Laboratory, Richland, Washington

7. PNNL-14670, *Vacuum Drying of Actual Transuranic Waste From Hanford Tanks*, J. M. Tingey, May 2004, Pacific Northwest National Laboratory, Richland, Washington

APPENDIX A

CH-TRUM Waste Shielding Analysis

Calculation Review Checklist

Calculation Reviewed: Dose Calculations for Transuranic (TRU) Waste Retrieval

Scope of Review: Entire Calculation, Note: Spreadsheet Verification is documented in SVF-227
(e.g., document section or portion of calculation)

Engineer/Analyst: R.L. Brown Date: July 7, 2004

Organizational Mgr: J.W. Hobbs Date: July 7, 2004

This document consists of 37 pages and the following attachments (if applicable):

N/A

Yes	No	NA*		
[✓]	[ ]	[ ]	1.	Analytical and technical approaches and results are reasonable and appropriate.
[✓]	[ ]	[ ]	2.	Necessary assumptions are reasonable, explicitly stated, and supported.
[✓]	[ ]	[ ]	3.	Ensure calculations that use software include a paper printout, microfiche, CD ROM, or other electronic file of the input data and identification to the computer codes and versions used, or provide alternate documentation to uniquely and clearly identify the exact coding and execution process.
[✓]	[ ]	[ ]	4.	Input data were checked for consistency with original source information.
[✓]	[ ]	[ ]	5.	For both qualitative and quantitative data, uncertainties are recognized and discussed.
[✓]	[ ]	[ ]	6.	Mathematical derivations were checked including dimensional consistency of results.
[✓]	[ ]	[ ]	7.	Calculations are sufficiently detailed such that a technically qualified person can understand the analysis without requiring outside information.
[✓]	[ ]	[ ]	8.	Software verification and validation are addressed adequately.
[✓]	[ ]	[ ]	9.	Limits/criteria/guidelines applied to the analysis results are appropriate and referenced. Limits/criteria/guidelines were checked against references.
[✓]	[ ]	[ ]	10.	Conclusions are consistent with analytical results and applicable limits.
[✓]	[ ]	[ ]	11.	Results and conclusions address all points in the purpose.
[✓]	[ ]	[ ]	12.	Referenced documents are retrievable or otherwise available.
[✓]	[ ]	[ ]	13.	The version or revision of each reference is cited.
[✓]	[ ]	[ ]	14.	The document was prepared in accordance with Attachment A, "Calculation Format and Preparation Instructions."
[✓]	[ ]	[ ]	15.	All checker comments have been dispositioned and the design media matches the calculations.

R.M. Pierson R.M. P. 07/20/04
Checker (Printed Name and Signature) Date

* If No or NA is chosen, an explanation must be provided on or attached to this form.

Section A-1
Dose Calculations for Transuranic (TRU) Waste Retrieval

1.0 OBJECTIVES/PURPOSE/SCOPE

This appendix provides the basis for the dose rate values contained in the tables presented in section A-2. These tables establish a correlation between product density and specific activity to external dose rates and provide a tool to approximate dose rates associated with contact handled TRUM waste for the following configurations:

- Single 55 Gallon Drum sides @ contact, 30 cm, and 2 meters
- TRUPACT-II Container with a 7 drum pallet as measured from the bottom @ contact, 30 cm, and 2 meters
- TRUPACT-II Container with a 7 drum pallet as measured from the side @ contact, 30 cm, and 2 meters
- Single SWB bottom @ contact, 30 cm, and 2 meter
- TRUPACT-II Container with 2 SWB, bottom @ contact, 30 cm, and 2 meter
- TRUPACT-II Container sides with 2 SWB @ contact, 30 cm, and 2 meter

The scope of the calculations contained in this document is limited to the configurations and applications described.

2.0 SUMMARY OF RELEVANT DOSE RATE REQUIREMENTS

Dose rates at contact (1.75"), thirty centimeters (cm) and at two meters (M) are critical to waste containers. The following summarizes the specific references for the distances and the relative dose rate criteria for each distance.

2.1 HNF-EP-0063 Rev. 10, Hanford Site Solid Waste Acceptance Criteria

Page 2-13 - Waste packages received by the Central Waste Complex (CWC) shall not exceed 1 milliSievert per hour (100 millirem per hour) at 30 centimeters (1 foot) from the waste package and 2 milliSieverts per hour (200 millirem per hour) at any point on the surface of the package.

2.2 DOE/WIPP-02-3122 Rev. 1, Contact-Handled Transuranic Waste Acceptance Criteria for the Waste Isolation Pilot Plant (WIPP)

Page 3-14 - *Acceptance Criterion.* The external radiation dose equivalent rate of individual payload containers shall be (less than or equal to) 200 mrem/hr at the surface. The external radiation dose equivalent rate of the TRUPACT-II and HalfPACT shall be (less than or equal to) 200 mrem/hr at the surface and (less than or equal to) 10 mrem/hr at 2 M.

Individual "Payload Containers" are defined by WIPP as being a 55-gallon **Drum,** standard waste box **(SWB),** or a ten drum overpack (TDOP), page C-4, DOE/WIPP-02-3122. For the purpose of the TRU dose evaluation we are only interested in the Drum and SWB.

The "Package" is defined by WIPP as being the TRUPACT-II or HalfPACT loaded with TRU waste payload containers; page C-4, DOE/WIPP-02-3122. For the purpose of this TRU dose evaluation we are interested in the evaluation of the TRUPACT-II with payload containers inside. The payload containers will consist of one of the three following:

- 7 pack of drums
- 2 SWBs direct filled with waste
- 2 SWBs containing 4 drums each

Note: The current project **assumption** is that the TRUPACT-II will be loaded with 7 empty (dunage) drums stacked upon the 7 Drums filled with waste, all inside the TRUPACT-II. Therefore only 7 drums are considered.

Table 1: Summary of the Relevant Dose Rate Criteria for Packages

	Contact 200 mR/hr	30 cm 100 mR/hr	2 M 10 mR/hr
Single 55 Gallon Drum	CWC/WIPP	CWC	
TRUPACT-II, 7 Drums	WIPP		WIPP
Single SWB	CWC/WIPP	CWC	
TRUPACT-II, 2 SWB	WIPP		WIPP

3.0 SUMMARY OF RESULTS

Tables that can be used to determine the dose rates for the select tanks are included in section A-2 of this appendix.

4.0 ASSUMPTIONS AND INPUT

4.1 General

The following assumptions were applied throughout this document.

- The primary gamma producing nuclide of concern is cesium 137 (^{137}Cs). The metastable barium-137 (^{137m}Ba) decay product of ^{137}Cs emits over 99% of the penetrating radiation from the waste material, excluding any Bremsstrahlung photons from associated beta decay.

- The contribution from Bremsstrahlung is tank specific based on Sr^{90} to Cs^{137} ratios and is determined as shown in section A-4 using data from the Best Basis Inventory (BBI) associated with each tank.
- The estimated ^{137}Cs concentrations in the final package are based on material density which is unknown.
- Contact dose rates were modeled at 1.75 in (3.9 cm) to reflect the active region of the ion chamber for the RO-20 instrument to record radiation readings on containers at both CH2MHILL and the CWC.
- The material makeup of 241-B-202 provides a representative approximation of the chemical makeup for the tanks under consideration.

4.2 Custom Material Files

Determining the appropriate material to model for the source (box or drum contents) is critical. The majority of radiation measured is a result of Buildup. Buildup, as calculated by MicroShield, is based on the effective atomic number of the waste form. And can vary by up to 50% for different materials. A review of the four most limiting tanks under consideration: B-201, B-202, T-104, and T-111 reveals that, although the waste forms are similar, subtle differences within the waste tanks under consideration can cause the final results to vary by as much as 30%.

Given the uncertainty of the materials in the tanks a review of the effects of the buildup for these four tanks was evaluated. The results of this evaluation determined that Tank 241-B-202 contains, as the top three components, Nitrate, Sodium and Bismuth. These chemicals are contained in approximately similar ratios as the other tanks being evaluated. Because of this, the material makeup of 241-B-202 was used as the custom material to model the source and will approximate the waste material for this evaluation.

The custom source material is approximated by the following table:

Table 2: Custom Source Material File

Weight Percent	Material
20.1 %	Water
13.5 %	Bismuth
24.5 %	Nitrate
15.6 %	Sodium
4.9 %	Manganese
3.9 %	Carbonate
5.1 %	Phosphate
4.8 %	Lanthanum
4.8 %	Iron
2.9 %	Silicon

4.3 Source Terms

Initial Process Engineering estimates of ^{137}Cs concentrations and density of source material for 20 weight % water are shown in the table below. This table can be used to approximate the dose rates on containers using the tables provided in section A-2. The table only reflects values for the 20 weight % water for the specific source material used in the model. Modification of the source material by the addition of sorbents can modify results and will require additional evaluation.

Table 3: Source Term Values for 20% Moisture

Tank	Estimated Product Density at 20 wt% water	Cs-137 Concentration (uCi/cc) at 20 wt% water	Bremsstrahlung Correction Factor	Decay Factor Corrected to June 1, 2005	Corrected for Decay and Bremsstrahlung (μCi/ml)
241-B-201	1.45	0.48	1.20	0.90	0.52
241-B-202	1.45	0.39	1.50	0.90	0.53
241-B-203	1.45	0.04	1.10	0.90	0.04
241-B-204	1.45	0.15	1.60	0.90	0.22
241-T-201	1.56	0.09	1.00	0.90	0.08
241-T-202	1.56	0.14	1.00	0.90	0.13
241-T-203	1.48	0.07	1.00	0.90	0.06
241-T-204	1.64	0.03	1.00	0.90	0.03
241-T-104	1.46	0.66	1.20	0.90	0.71
241-T-110	1.46	0.07	1.00	0.90	0.07
241-T-111 upper segments	1.46	1.78	1.50	1.00	2.67
241-T-111 Average	1.46	0..51	1.50	0.90	0.69

The Bremsstrahlung correction factor for each tank is shown below, along with the corresponding ^{90}Sr: ^{137}Cs ratios used to develop the factor.

Table 4: Bremsstrahlung Correction Factor

Tank	Cesium (Ci)	Strontium (Ci)	Ratio (Sr/Cs)	Bremsstrahlung Correction Factor
241-B-201	2.11E+01	2.67E+02	13:1	1.2
241-B-202	1.11E+01	4.37E+02	39:1	1.5
241-B-203	1.92E+00	1.73E+01	9:1	1.1
241-B-204	6.67E+00	3.15E+02	47:1	1.6
241-T-104	2.57E+02	3.38E+03	13:1	1.2
241-T-110	2.78E+01	5.12E+01	2:1	1.0
241-T-111	2.29E+02	9.23E+03	40:1	1.5
241-T-201	4.40E+00	1.13E+01	3:1	1.0
241-T-202	2.61E+00	2.30E-01	0.1:1	1.0
241-T-203	2.51E+00	4.07E-01	0.2:1	1.0
241-T-204	1.24E+00	7.32E-01	0.6:1	1.0

For more information on the development of this table, see section A-4: Development of Bremsstrahlung Correction Factors.

4.4 Model Input

The following reflects the input used to define the MicroShield models used to calculate the dose rates.

4.4.1 55 Gallon Drum

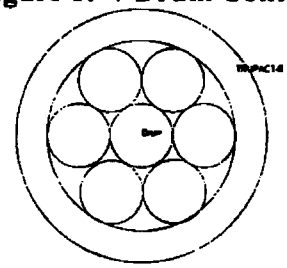
The 55 gallon drum is modeled as completely full of waste. The following information was used in the model:

Drum Height	33.3 in (84.6 cm)
Drum Radius	11.25 Inches
Drum Wall Thickness	0.053 in (.0315 cm)
Material	Iron
Density:	7.86 gm/cc

4.4.2 TRUPACT-II Container with a 7 Drum Pallet - Side

This model considered only the dose rates from the three drums closest to the side of the TRUPACT-II. No significant contribution to dose rates came from drums in the shadow of these three drums. Figure 1, TRUPACT-II Drum Assembly shows the drum configuration.

Figure 1: 7 Drum Configurations



The following information was used in the model:

Inner Containment Vessel (ICV) Shell

Material:	DT-304 Stainless Steel
Density:	7.97 gm/cc
Thickness:	0.25 in (0.636 cm)

ICV-Outer Containment Vessel (OCV) Shell Air Gap

Material:	Air
Density	0.00122 gm/cc
Thickness:	0.25 in (0.636 cm)

OCV Shell

Material:	DT-304 Stainless Steel
Density	7.97 gm/cc
Thickness	0.1875 in (0.476 cm)

Polyurethane Foam

Material:	Polyurethane Foam and Ceramic Fiber
Density	0.136 gm/cc
Thickness	10.125 in (25.7 cm)

Note: There are two layers of ceramic fiber modeled as foam and included in the foam layer dimensions.

Outer Containment Vessel (OCV) Body

Material:	DT-304 Stainless Steel
Density:	7.97 gm/cc
Thickness:	0.25 in (0.636 cm)

4.4.3 TRUPACT-II Container with 7 Drum Pallet- Bottom

This model considered the maximum dose rate from the center drum as the primary contributor and the contribution from the six surrounding drums as secondary contributors. Figure 1 approximates this configuration.

Inner Containment Vessel (ICV) Lower Spacer Assembly

Material:	Honeycombed Aluminum
Density:	0.058 gm/cc
Thickness:	11.313 in (28.7 cm)

Inner Containment Vessel (ICV) Shell

Material:	DT-304 Stainless Steel
Density:	7.97 gm/cc
Thickness:	0.25 in (0.636 cm)

ICV-Outer Containment Vessel (OCV) Shell Air Gap

Material:	Air
Density	0.00122 gm/cc
Thickness:	0.25 in (0.636 cm)

OCV Shell

Material:	DT-304 Stainless Steel
Density	7.97 gm/cc
Thickness	0.1875 in (0.476 cm)

Polyurethane Foam

Material:	Polyurethane Foam and Ceramic Fiber
Density	0.136 gm/cc
Thickness	9 in (22.9 cm)

Note: There are two layers of ceramic fiber that line the foam that are modeled as foam and included in the foam layer dimensions.

Outer Containment Vessel (OCV) Body

Material:	DT-304 Stainless Steel
Density:	7.97 gm/cc
Thickness:	0.25 in (0.636 cm)

Note: The Outer Containment Vessel contains two fork truck slots that are not considered when determining distances for dose rates.

4.4.4 Single SWB bottom

The following dimensions used for model:

Standard Waste Box	
Height (Waste in Box)	20 in (50.8 cm)
Width	52 in (132cm)
Length	68 in (173 cm)
Wall Thickness	0.135 in (XX cm)
Material	Iron
Density	7.86 gm/cc

The SWB box bottom is the limiting geometry due to the large exposed plane at the bottom of this waste container. The boxes are assumed to be filled with 20" of waste. At current weight limitations, no additional material beyond a 20" fill will be possible. The model assumes the SWB to be a rectangle and ignores the end curvature. This results in slight conservatisms within the model, but does not significantly affect the calculated dose rate

4.4.5 TRUPACT-II Container with 2 SWB, bottom

The TRUPACT-II with two stacked SWBs, will have a relatively insignificant contribution from the second SWB. Therefore, for the purposes of this evaluation the upper box was not considered in the calculation.

The material information for this model is the same as for the model used in Section 4.3.3, TRUPACT-II Container with 7 drum pallet - Bottom

4.4.6 TRUPACT-II Container sides with 2 SWB

With two SWBs in the TRUPACT-II, both boxes contribute to the dose rates and are therefore both considered.

The material information for this model is the same as for the model used in Section 4.3.2, TRUPACT-II Container with a 7 drum pallet - Side

5.0 METHODOLOGY

MicroShield software version 6.02 was used to determine the exposure rates for all described configurations. The output of the individual MicroShield runs are used to develop the tables presented in attachment A-2 that represent the anticipated dose rates for variations in density and source term. Software Verification and Validation runs for MicroShield, version 6.02, release date 2003, were performed on 5/1/2004 on USDOE

computer WC87460. Microsoft Excel was used for developing the dose rate versus density tables. The version of Excel available on the Hanford HLAN was used and spreadsheet verification was performed and documented on a Spread Sheet Verification Form. The spreadsheet verification was documented under the verification number SVF-227.

6.0 CONCLUSIONS

No distinct conclusions are drawn from the data presented, as that is beyond the scope of the calculations.

7.0 REFERENCES

1. Internal Memo, M. E. Johnson to A. R. Tedeschi, 7G330-MEJ-04-001, Density Estimate for Packaged Transuranic Tank Waste, March 25, 2004
2. Internal Memo, M. E. Johnson to A. R. Tedeschi, 7G330-MEJ-04-004, Cesium-137 Concentrations in Transuranic Waste Tanks, April 28, 2004
3. Internal Memo, M. E. Johnson to A. R. Tedeschi, 7G330-MEJ-04-005, Change in Cesium-137 Concentrations Based on Water Content for Transuranic Tank Wastes, April 29, 2004
4. RPP-15219, Characteristics of Transuranic Wastes in the Hanford B and T Tank Farms, April 4, 2003.
5. HNF-EP-0063, Hanford Site Solid Waste Acceptance Criteria, Revision 10, 2004, February 17, 2004.
6. DOE/WIPP-02-3122, Contact-Handled Transuranic Waste Acceptance Criteria for the Waste Isolation Pilot Plant, Revision 1, March 1, 2004.
7. TRUPACT-II Packaging SAR Drawing, 2077-500SNP, Revision 1, March 11, 2004.
8. TRAMPAC Document, Appendix 2.1 November 2000
9. TRAMPAC SAR, Appendix 1.3.4, April 2000,
10. Handbook of Radiation Measurement and Protection, Section A, Volume I, Allen Brodsky, 1978
11. Drawing 165-F-001-W1, Standard Waste Box Assembly, Revision U, April 14, 2003.

Section A-2
Dose Rate Verses Density Tables

Originator: R.L. Brown Date: July 7, 2004

Checker: R. M Pierson Date: July 7, 2004

Organizational Manager: J. W. Hobbs Date: July 7, 2004

Single 55 Gallon Drum Side (Limiting- Contact)

Density	0.3 uCi/ml			0.4 uCi/ml			0.5 uCi/ml			0.6 uCi/ml			0.7 uCi/ml			0.8 uCi/ml			0.9 uCi/ml			1 uCi/ml			1.1 uCi/ml			1.2uCi/ml			1.3 uCi/ml			1.4 uCi/ml			1.5 uCi/ml			1.6 uCi/ml		
	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm
1	81	2	28	107	3	38	134	3	47	161	4	57	186	5	66	215	5	76	242	6	85	269	6	94	296	7	104	322	8	113	349	8	123	376	9	132	403	10	142	430	10	151
1.1	78	2	27	100	2	35	125	3	44	150	4	53	175	4	62	200	5	71	225	5	80	250	6	89	275	7	97	300	7	106	326	8	115	351	8	124	376	9	133	401	10	142
1.2	70	2	25	94	2	33	117	3	42	141	3	50	164	4	58	187	5	67	211	5	75	234	6	83	258	6	91	281	7	100	304	7	108	328	8	116	351	8	125	375	9	133
1.3	66	2	24	88	2	31	110	3	39	132	3	47	154	4	55	176	4	63	198	5	71	220	5	78	242	6	86	264	6	94	286	7	102	308	7	110	330	8	118	352	8	125
1.4	62	1	22	83	2	30	103	2	37	124	3	44	145	3	52	166	4	59	186	4	67	207	5	74	227	5	81	248	6	89	269	6	96	289	7	104	310	7	111	331	8	118
1.5	59	1	21	78	2	28	98	2	35	117	3	42	137	3	49	156	4	56	176	4	63	195	5	70	215	5	77	234	6	84	254	6	91	273	7	98	293	7	105	312	7	112
1.6	55	1	20	74	2	27	92	2	33	111	3	40	129	3	47	148	4	53	166	4	60	184	4	67	203	5	73	221	5	80	240	6	86	258	6	93	277	7	100	295	7	106
1.7	52	1	19	70	2	25	87	2	32	105	3	38	122	3	44	140	3	51	157	4	57	175	4	63	192	5	70	210	5	76	227	5	82	245	6	89	262	6	95	280	7	101
1.8	50	1	18	66	2	24	83	2	30	100	2	36	116	3	42	133	3	48	150	4	54	166	4	60	183	4	66	199	5	72	216	5	78	233	6	84	249	6	90	266	6	96
1.9	47	1	17	63	2	23	79	2	29	95	2	35	111	3	40	127	3	46	142	3	52	158	4	58	174	4	63	190	5	69	206	5	75	221	5	81	237	6	86	253	6	92
2	45	1	17	60	1	22	76	2	28	91	2	33	106	3	39	121	3	44	136	3	50	151	4	55	166	4	61	181	4	66	196	5	72	211	5	77	227	5	83	242	6	88

TRUPACT-II, 7 ea 55 Gallon Drum Bottom (Limiting 2 M)

Density	0.3 uCi/ml			0.4 uCi/ml			0.5 uCi/ml			0.6 uCi/ml			0.7 uCi/ml			0.8 uCi/ml			0.9 uCi/ml			1 uCi/ml			1.1 uCi/ml			1.2uCi/ml			1.3 uCi/ml			1.4 uCi/ml			1.5 uCi/ml			1.6 uCi/ml		
	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm
1	18	2	12	24	3	16	30	4	21	35	5	25	41	5	29	47	6	33	53	7	37	59	8	41	66	9	45	71	9	49	77	10	54	83	11	58	89	12	62	95	13	66
1.1	16	2	11	22	3	15	27	4	19	33	4	23	38	5	27	44	6	31	49	6	34	55	7	38	60	8	42	66	9	46	71	9	50	77	10	54	82	11	58	88	12	61
1.2	15	2	11	21	3	14	26	3	18	31	4	21	36	5	25	41	5	28	46	6	32	51	7	36	56	7	39	62	8	43	67	9	46	72	9	50	77	10	53	82	11	57
1.3	14	2	10	19	2	13	24	3	17	29	4	20	34	4	23	38	5	27	43	6	30	48	6	33	53	7	37	58	7	40	62	8	43	67	9	47	72	9	50	77	10	53
1.4	14	2	9	18	2	13	23	3	16	27	3	19	32	4	22	36	5	25	41	5	28	45	6	31	50	6	34	54	7	38	59	8	41	63	8	44	68	9	47	72	9	50
1.5	13	2	9	17	2	12	21	3	15	26	3	18	30	4	21	34	4	24	38	5	27	43	6	30	47	6	32	51	7	35	56	7	38	60	8	41	64	8	44	68	9	47
1.6	12	2	8	16	2	11	20	3	14	24	3	17	28	4	20	32	4	23	36	5	25	40	5	28	44	6	31	48	6	34	52	7	37	56	7	39	60	8	42	64	8	45
1.7	11	1	8	15	2	11	19	2	13	23	3	16	27	3	19	31	4	21	34	4	24	38	5	26	42	5	29	46	6	32	50	6	34	53	7	37	57	7	40	61	8	42
1.8	11	1	8	15	2	10	18	2	13	22	3	15	25	3	18	29	4	20	33	4	23	36	5	25	40	5	28	44	6	30	47	6	33	51	6	35	54	7	38	58	7	40
1.9	10	1	7	14	2	10	17	2	12	21	3	14	24	3	17	28	4	19	31	4	22	35	4	24	38	5	26	42	5	29	45	6	31	48	6	34	52	7	36	55	7	38
2	10	1	7	13	2	9	17	2	11	20	3	14	23	3	16	26	3	18	30	4	21	33	4	23	36	5	25	40	5	27	43	5	30	46	6	32	50	6	34	53	7	37

7 ea 55 Gallon Drum Side in a TRUPACTII (Limiting 2 M)

Density	0.3 uCi/ml			0.4 uCi/ml			0.5 uCi/ml			0.6 uCi/ml			0.7 uCi/ml			0.8 uCi/ml			0.9 uCi/ml			1 uCi/ml			1.1 uCi/ml			1.2uCi/ml			1.3 uCi/ml			1.4 uCi/ml			1.5 uCi/ml			1.6 uCi/ml		
	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm
1	18	2	11	24	2	14	29	3	18	36	3	21	41	4	25	47	4	28	53	5	32	59	6	35	65	6	39	71	7	43	77	7	46	83	8	50	88	8	53	94	9	57
1.1	16	2	10	22	2	13	27	3	17	33	3	20	38	4	23	44	4	26	49	5	30	55	5	33	60	6	36	66	6	40	71	7	43	77	7	46	82	8	50	88	8	53
1.2	15	1	9	20	2	12	26	2	15	31	3	18	36	3	22	41	4	25	46	4	28	51	5	31	56	5	34	61	6	37	67	6	40	72	7	43	77	7	46	82	8	49
1.3	14	1	9	19	2	12	24	2	14	29	3	17	34	3	20	38	4	23	43	4	26	48	5	29	53	5	32	58	5	35	62	6	38	67	6	40	72	7	43	77	7	46
1.4	14	1	8	18	2	11	23	2	14	27	3	16	32	3	19	36	3	22	41	4	24	45	4	27	50	5	30	54	5	33	59	6	35	63	6	38	68	6	41	72	7	43
1.5	13	1	8	17	2	10	21	2	13	25	2	15	30	3	18	34	3	20	38	4	23	42	4	26	47	4	28	51	5	31	55	5	33	60	6	36	64	6	38	68	6	41
1.6	12	1	7	16	2	10	20	2	12	24	2	15	28	3	17	32	3	19	36	3	22	40	4	24	44	4	27	48	5	29	52	5	31	56	5	34	60	6	36	64	6	39
1.7	11	1	7	15	1	9	19	2	11	23	2	14	27	2	16	30	3	18	34	3	21	38	4	23	42	4	25	46	4	28	50	5	30	53	5	32	57	5	34	61	6	37
1.8	11	1	7	14	1	9	18	2	11	22	2	13	25	2	15	29	3	17	33	3	20	36	3	22	40	4	24	43	4	26	47	4	28	51	5	31	54	5	33	58	5	35
1.9	11	1	7	15	1	9	19	2	11	22	2	13	26	2	16	30	3	18	34	3	20	37	3	22	41	4	25	45	4	27	49	4	29	52	5	31	56	5	34	60	5	36
2	10	1	6	13	1	8	16	2	10	20	2	12	23	2	14	26	2	16	30	3	18	33	3	20	36	3	22	40	4	24	43	4	26	46	4	28	49	5	30	53	5	32

SWB in TRUPACT II Bottom (Limiting 2 M)

Density	0.3 uCi/ml			0.4 uCi/ml			0.5 uCi/ml			0.6 uCi/ml			0.7 uCi/ml			0.8 uCi/ml			0.9 uCi/ml			1 uCi/ml			1.1 uCi/ml			1.2uCi/ml			1.3 uCi/ml			1.4 uCi/ml			1.5 uCi/ml			1.6 uCi/ml		
	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm
1	17	3	12	22	4	17	28	4	21	33	5	25	39	6	29	44	7	33	50	8	37	56	9	41	61	10	46	67	11	50	72	11	54	78	12	58	83	13	62	89	14	66
1.1	16	2	11	21	3	15	26	4	19	31	5	23	36	6	27	41	7	31	46	7	34	51	8	38	56	9	42	62	10	46	67	11	50	72	11	54	77	12	57	82	13	61
1.2	14	2	11	19	3	14	24	4	18	29	5	21	33	5	25	38	6	28	43	7	32	48	8	36	52	8	39	57	9	43	62	10	46	67	11	50	71	11	53	76	12	57
1.3	13	2	10	18	3	13	22	4	17	27	4	20	31	5	23	36	6	27	40	6	30	44	7	33	49	8	37	53	9	40	58	9	43	62	10	47	67	11	50	71	11	53
1.4	12	2	9	17	3	12	21	3	16	25	4	19	29	5	22	33	5	25	37	6	28	42	7	31	46	7	34	50	8	37	54	9	41	58	9	44	62	10	47	67	11	50
1.5	12	2	9	16	3	12	20	3	15	23	4	18	27	4	21	31	5	23	35	6	26	39	6	29	43	7	32	47	8	35	51	8	38	55	9	41	59	9	44	63	10	47
1.6	11	2	8	15	2	11	18	3	14	22	4	17	26	4	19	29	5	22	33	5	25	37	6	28	41	6	30	44	7	33	48	8	36	52	8	39	55	9	42	59	9	44
1.7	10	2	8	14	2	10	17	3	13	21	3	16	24	4	18	28	4	21	31	5	24	35	6	26	38	6	29	42	7	31	45	7	34	49	8	37	52	8	39	56	9	42
1.8	10	2	7	13	2	10	17	3	12	20	3	15	23	4	17	26	4	20	30	5	22	33	5	25	36	6	27	40	6	30	43	7	32	46	7	35	50	8	37	53	8	40
1.9	9	2	7	13	2	9	16	3	12	19	3	14	22	4	17	25	4	19	28	5	21	31	5	24	35	6	26	38	6	28	41	7	31	44	7	33	47	8	35	50	8	38
2	9	1	7	12	2	9	15	2	11	18	3	14	21	3	16	24	4	18	27	4	20	30	5	23	33	5	25	36	6	27	39	6	29	42	7	32	45	7	34	48	8	36

2 ea 20" SWB, Sides in TrupactII (Limiting 2 M)

Density	0.3 uCi/ml			0.4 uCi/ml			0.5 uCi/ml			0.6 uCi/ml			0.7 uCi/ml			0.8 uCi/ml			0.9 uCi/ml			1 uCi/ml			1.1 uCi/ml			1.2uCi/ml			1.3 uCi/ml			1.4 uCi/ml			1.5 uCi/ml			1.6 uCi/ml		
	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm
1	18	3	13	25	4	18	31	5	22	37	6	27	43	7	31	49	7	35	55	8	40	61	9	44	67	10	49	74	11	53	80	12	58	86	13	62	92	14	66	98	15	71
1.1	17	3	12	23	3	16	28	4	20	34	6	25	40	6	29	45	7	33	51	8	37	57	9	41	62	9	45	68	10	49	74	11	53	79	12	57	85	13	61	91	14	65
1.2	16	2	11	21	3	15	26	4	19	32	5	23	37	6	27	42	6	30	47	7	34	53	8	38	58	9	42	63	10	46	68	10	49	74	11	53	79	12	57	84	13	61
1.3	15	2	11	20	3	14	25	4	18	30	4	21	34	5	25	39	6	28	44	7	32	49	7	35	54	8	39	59	9	43	64	10	46	69	10	50	74	11	53	79	12	57
1.4	14	2	10	18	3	13	23	3	17	28	4	20	32	5	23	37	6	27	42	6	30	46	7	33	51	8	37	55	8	40	60	9	43	65	10	47	69	10	50	74	11	53
1.5	13	2	9	17	3	13	22	3	16	26	4	19	30	5	22	35	5	25	39	6	28	43	7	31	48	7	34	52	8	38	56	8	41	61	9	44	65	10	47	70	10	50
1.6	12	2	9	16	2	12	21	3	15	25	4	18	29	4	21	33	5	24	37	6	27	41	6	30	45	7	32	49	7	35	53	8	38	57	9	41	62	9	44	66	10	47
1.7	12	2	8	16	2	11	19	3	14	23	3	17	27	4	20	31	5	22	35	5	25	39	6	28	43	6	31	47	7	34	51	8	36	54	8	39	58	9	42	62	9	45
1.8	11	2	8	15	2	11	18	3	13	22	3	16	26	4	19	30	4	21	33	5	24	37	6	27	41	6	29	44	7	32	48	7	35	52	8	37	55	8	40	59	9	43
1.9	11	2	8	14	2	10	18	3	13	21	3	15	25	4	18	28	4	20	32	5	23	35	5	25	39	6	28	42	6	30	46	7	33	49	7	35	53	8	38	56	8	41
2	10	2	7	13	2	10	17	3	12	20	3	14	24	4	17	27	4	19	30	5	22	34	5	24	37	6	27	40	6	29	44	7	31	47	7	34	50	8	36	54	8	39

SWB, Bottom (Limiting 30 cm)

Density	0.3 uCi/ml			0.4 uCi/ml			0.5 uCi/ml			0.6 uCi/ml			0.7 uCi/ml			0.8 uCi/ml			0.9 uCi/ml			1 uCi/ml			1.1 uCi/ml			1.2uCi/ml			1.3 uCi/ml			1.4 uCi/ml			1.5 uCi/ml			1.6 uCi/ml		
	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm	Contact	2 M	30 cm
1	108	9	75	144	12	100	181	15	125	217	18	150	253	21	175	289	25	200	325	28	226	361	31	251	397	34	276	433	37	301	470	40	326	506	43	351	542	46	376	578	49	401
1.1	99	9	70	132	11	93	166	14	116	199	17	139	232	20	163	265	23	186	298	26	209	331	28	232	364	31	255	397	34	279	430	37	302	463	40	325	497	43	348	530	46	372
1.2	92	8	66	122	11	88	153	13	108	183	16	130	214	18	151	244	21	173	275	24	194	306	27	216	336	29	238	366	32	259	397	35	281	427	37	302	458	40	324	488	42	346
1.3	85	7	61	113	10	81	141	12	101	170	15	121	198	17	141	226	20	161	254	22	182	283	25	202	311	27	222	339	30	242	368	32	262	396	35	282	424	37	303	452	40	323
1.4	79	7	57	105	9	76	132	12	95	158	14	113	184	16	132	211	19	151	237	21	170	263	23	188	290	26	208	316	28	227	342	30	246	368	33	265	395	35	284	421	37	303
1.5	74	7	53	98	9	71	123	11	89	148	13	107	172	15	124	197	18	142	221	20	160	246	22	178	271	24	195	295	26	213	320	28	231	345	31	249	369	33	267	394	35	284
1.6	69	6	50	92	8	67	116	10	84	139	12	101	162	15	117	185	17	134	208	19	151	231	21	168	254	23	185	277	25	201	300	27	218	324	29	235	347	31	252	370	33	268
1.7	66	6	48	87	8	64	109	10	79	131	12	95	152	14	111	174	16	127	196	18	143	218	20	159	239	22	175	261	24	191	283	25	206	302	28	222	327	30	238	348	32	254
1.8	62	6	45	82	7	60	103	9	75	123	11	90	144	13	106	165	15	121	185	17	136	206	19	151	226	21	166	247	22	181	267	24	196	288	26	211	309	28	226	329	30	241
1.9	59	5	43	78	7	57	98	9	72	117	11	86	137	12	100	156	14	115	176	16	129	195	18	143	215	20	158	234	21	172	254	23	186	273	25	201	293	27	215	312	29	229
2	56	5	41	74	7	55	93	9	68	111	10	82	130	12	96	148	14	109	167	15	123	185	17	137	204	19	150	222	20	164	241	22	178	259	24	191	278	26	205	296	27	219

Section A-3
MicroShield Runs

Originator: R.L. Brown Date: July 7, 2004

Checker: R. M Pierson Date: July 7, 2004

Organizational Manager: J. W. Hobbs Date: July 7, 2004

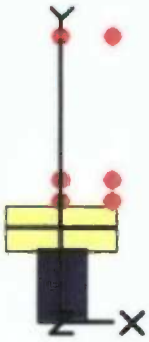
MicroShield v6.02 (6.02-00127)
CH2M_Hill_Hanford_Group_Inc.

Page	:1	**File Ref**	:Typical 7 Drum on Bottom of TRUPACT-II Model
DOS File	:7b1.0.ms6		
Run Date	: May 17, 2004	**Date**	: 5/17/2004
Run Time	: 8:03:51 AM	**By**	: R. L. Brown
Duration	: 00:00:01	**Checked**	: R. M. Pierson

Case Title: Tru Drum Bottom
Description: Trupac II bottom of 7 drum array
Geometry: 8 - Cylinder Volume - End Shields

Source Dimensions:

Height	84.455 cm	(2 ft 9.3 in)
Radius	28.575 cm	(11.3 in)



Dose Points

A	X	Y	Z
# 1	0 cm	1.43e+02 cm	0 cm
	0.0 in	4 ft 8.4 in	0.0 in
# 2	0 cm	339 cm	0 cm
	0.0 in	11 ft 1.5 in	0.0 in
# 3	0 cm	169.2656 cm	0 cm
	0.0 in	5 ft 6.6 in	0.0 in
# 4	60.96 cm	1.43e+02 cm	0 cm
	2 ft	4 ft 8.4 in	0.0 in
# 5	60.96 cm	339 cm	0 cm
	2 ft	11 ft 1.5 in	0.0 in
# 6	60.96 cm	169.2656 cm	0 cm
	2 ft	5 ft 6.6 in	0.0 in

Shields

Shield N	Dimension	Material	Density
Source	1.32e+04 in³	B-202 TRU Custom	1
Shield 1	.08 in	Aluminum	2.7
Shield 2	11.313 in	Al Hcomb	0.058
Shield 3	.25 in	304 SST	7.97
Shield 4	.25 in	Air	0.00122
Shield 5	.188 in	304 SST	7.97
Shield 6	9.0 in	TRUPOLY	0.136
Shield 7	.25 in	304 SST	7.97
Air Gap		Air	0.00122
Wall Clad	.053 in	Iron	7.86
Top Clad	.053 in	Iron	7.86
Immersion		Al Hcomb	0.058

Source Input : Grouping Method - Actual Photon Energies

Nuclide	curies	becquerels	μCi/cm³	Bq/cm³
Ba-137m	2.0495e-001	7.5830e+009	9.4600e-001	3.5002e+004
Cs-137	2.1664e-001	8.0158e+009	1.0000e+000	3.7000e+004

Originator: R.L. Brown Date: July 7, 2004

Checker: R. M Pierson Date: July 7, 2004

Organizational Manager: J. W. Hobbs Date: July 7, 2004

Buildup : The material reference is - Source
This buildup reference material is a mixed material with a high atomic number element (83). Buildup Factors less than and somewhat greater than 91 keV may be incorrect. Please understand your results.

Integration Parameters

Radial	20
Circumferential	10
Y Direction (axial)	10

Results - Dose Point # 1 - (0,56.384,0) in

Energy MeV	Activity Photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0045	7.872e+07	0.000e+00	2.834e-26	0.000e+00	1.943e-26
0.0318	1.570e+08	4.372e-46	4.774e-25	3.642e-48	3.976e-27
0.0322	2.896e+08	2.379e-44	8.950e-25	1.914e-46	7.203e-27
0.0364	1.054e+08	4.482e-32	3.884e-25	2.547e-34	2.207e-27
0.6616	6.823e+09	2.031e+03	7.676e+03	3.938e+00	1.488e+01
Totals	**7.454e+09**	**2.031e+03**	**7.676e+03**	**3.938e+00**	**1.488e+01**

Results - Dose Point # 2 - (0,1.33e+02,0) in

Energy MeV	Activity Photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0045	7.872e+07	0.000e+00	2.884e-27	0.000e+00	1.977e-27
0.0318	1.570e+08	2.005e-46	4.857e-26	1.670e-48	4.046e-28
0.0322	2.896e+08	1.064e-44	9.106e-26	8.560e-47	7.328e-28
0.0364	1.054e+08	1.533e-32	3.952e-26	8.708e-35	2.245e-28
0.6616	6.823e+09	1.564e+02	6.136e+02	3.032e-01	1.190e+00
Totals	**7.454e+09**	**1.564e+02**	**6.136e+02**	**3.032e-01**	**1.190e+00**

Results - Dose Point # 3 - (0,66.64,0) in

Energy MeV	Activity Photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0045	7.872e+07	0.000e+00	1.679e-26	0.000e+00	1.151e-26
0.0318	1.570e+08	4.397e-46	2.828e-25	3.663e-48	2.356e-27
0.0322	2.896e+08	2.390e-44	5.302e-25	1.923e-46	4.267e-27
0.0364	1.054e+08	4.409e-32	2.301e-25	2.505e-34	1.307e-27
0.6616	6.823e+09	1.143e+03	4.337e+03	2.216e+00	8.408e+00
Totals	**7.454e+09**	**1.143e+03**	**4.337e+03**	**2.216e+00**	**8.408e+00**

Results - Dose Point # 4 - (24,56.384,0) in

Energy MeV	Activity Photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0045	7.872e+07	0.000e+00	1.932e-26	0.000e+00	1.325e-26
0.0318	1.570e+08	6.578e-55	3.255e-25	5.479e-57	2.711e-27
0.0322	2.896e+08	6.536e-53	6.102e-25	5.260e-55	4.911e-27
0.0364	1.054e+08	2.272e-38	2.648e-25	1.291e-40	1.505e-27
0.6616	6.823e+09	8.387e+02	3.815e+03	1.626e+00	7.396e+00
Totals	**7.454e+09**	**8.387e+02**	**3.815e+03**	**1.626e+00**	**7.396e+00**

Originator: R.L. Brown Date: July 7, 2004

Checker: R. M Pierson Date: July 7, 2004

Organizational Manager: J. W. Hobbs Date: July 7, 2004

Results - Dose Point # 5 - (24,1.33e+02,0) in

Energy MeV	Activity Photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0045	7.872e+07	0.000e+00	2.769e-27	0.000e+00	1.898e-27
0.0318	1.570e+08	1.302e-48	4.664e-26	1.084e-50	3.885e-28
0.0322	2.896e+08	7.825e-47	8.743e-26	6.298e-49	7.036e-28
0.0364	1.054e+08	3.417e-34	3.795e-26	1.941e-36	2.156e-28
0.6616	6.823e+09	1.425e+02	5.719e+02	2.763e-01	1.109e+00
Totals	**7.454e+09**	**1.425e+02**	**5.719e+02**	**2.763e-01**	**1.109e+00**

Results - Dose Point # 6 - (24,66.64,0) in

Energy MeV	Activity Photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0045	7.872e+07	0.000e+00	1.334e-26	0.000e+00	9.140e-27
0.0318	1.570e+08	9.868e-52	2.246e-25	8.220e-54	1.871e-27
0.0322	2.896e+08	7.404e-50	4.211e-25	5.959e-52	3.389e-27
0.0364	1.054e+08	2.278e-36	1.827e-25	1.294e-38	1.038e-27
0.6616	6.823e+09	6.576e+02	2.823e+03	1.275e+00	5.472e+00
Totals	**7.454e+09**	**6.576e+02**	**2.823e+03**	**1.275e+00**	**5.472e+00**

Originator: R.L. Brown Date: July 7, 2004

Checker: R. M Pierson Date: July 7, 2004

Organizational Manager: J. W. Hobbs Date: July 7, 2004

Table 1: Output of MicroShield Runs for 7 Drums, Bottom of TRUPACT-II

File Name	Point	Dose Rate (mR/hr)	File Name	Point	Dose Rate (mR/hr)
7b1.0.ms6	1	14.9	7b1.6.ms6	1	10.8
	2	1.2		2	0.8
	3	8.4		3	6.0
	4	7.4		4	4.9
	5	1.1		5	0.7
	6	5.5		6	3.7
7b1.1.ms6	1	13.9	7b1.7.ms6	1	9.9
	2	1.1		2	0.7
	3	7.8		3	5.5
	4	6.8		4	4.7
	5	1.0		5	0.7
	6	5.1		6	3.5
7b1.2.ms6	1	13.0	7b1.8.ms6	1	9.4
	2	1.0		2	0.7
	3	7.3		3	5.2
	4	6.4		4	4.5
	5	0.9		5	0.7
	6	4.7		6	3.3
7b1.3.ms6	1	12.2	7b1.9.ms6	1	9.0
	2	0.9		2	0.7
	3	6.8		3	5.0
	4	6.0		4	4.3
	5	0.9		5	0.6
	6	4.4		6	3.2
7b1.4.ms6	1	11.6	7b2.0.ms6	1	8.7
	2	0.9		2	0.6
	3	6.4		3	4.8
	4	5.6		4	4.1
	5	0.8		5	0.6
	6	4.1		6	3.0
7b1.5.ms6	1	10.9			
	2	0.8			
	3	6.1			
	4	5.3			
	5	0.8			
	6	3.9			

Originator: R.L. Brown Date: July 7, 2004

Checker: R. M Pierson Date: July 7, 2004

Organizational Manager: J. W. Hobbs Date: July 7, 2004

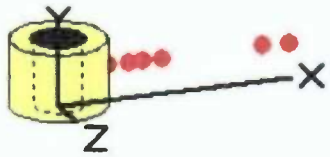
MicroShield v6.02 (6.02-00127)
CH2M_Hill_Hanford_Group_Inc.

Page	:1	**File Ref**	: Typical 7 Drum on Bottom of TRUPACT-II Model
DOS File	:7dp1.0.ms6		
Run Date	: May 17, 2004	**Date**	: 5/17/2004
Run Time	: 1:20:45 PM	**By**	: R. L. Brown
Duration	: 00:00:02	**Checked**	: R. M. Pierson

Case Title: TRUPACT-II 7 Drum
Description: Evaluation of Drum Bottom Loading on Trupact II
Geometry: 7 - Cylinder Volume - Side Shields

Source Dimensions:

Height	84.455 cm	(2 ft 9.3 in)
Radius	28.575 cm	(11.3 in)



Dose Points

A	X	Y	Z
# 1	6.38e+01 cm	41.91 cm	0 cm
	2 ft 1.1 in	1 ft 4.5 in	0.0 in
# 2	259 cm	41.91 cm	0 cm
	8 ft 6.0 in	1 ft 4.5 in	0.0 in
# 3	89.82964 cm	41.91 cm	0 cm
	2 ft 11.4 in	1 ft 4.5 in	0.0 in
# 4	108.204 cm	41.91 cm	0 cm
	3 ft 6.6 in	1 ft 4.5 in	0.0 in
# 5	294.386 cm	41.91 cm	0 cm
	9 ft 7.9 in	1 ft 4.5 in	0.0 in
# 6	131.191 cm	41.91 cm	0 cm
	4 ft 3.7 in	1 ft 4.5 in	0.0 in

Shields

Shield N	Dimension	Material	Density
Source	1.32e+04 in³	B-202 TRU Custom	1
Shield 1	1.0 in	Air	0.00122
Shield 2	.25 in	304 SST	7.97
Shield 3	.25 in	Air	0.00122
Shield 4	.188 in	304 SST	7.97
Shield 5	10.125 in	TRUPOLY	0.136
Shield 6	.25 in	304 SST	7.97
Transition		Air	0.00122
Air Gap		Air	0.00122
Wall Clad	.053 in	Iron	7.86
Top Clad	.053 in	Iron	7.86

Source Input : Grouping Method - Actual Photon Energies

Nuclide	curies	becquerels	µCi/cm³	Bq/cm³
Ba-137m	2.0340e-001	7.5260e+009	9.3889e-001	3.4739e+004
Cs-137	2.1502e-001	7.9556e+009	9.9248e-001	3.6722e+004

Originator: R.L. Brown Date: July 7, 2004

Checker: R. M Pierson Date: July 7, 2004

Organizational Manager: J. W. Hobbs Date: July 7, 2004

Buildup : The material reference is - Source
This buildup reference material is a mixed material with a high atomic number element (83). Buildup Factors less than and somewhat greater than 91 keV may be incorrect. Please understand your results.

Integration Parameters

Radial	10
Circumferential	10
Y Direction (axial)	20

Results - Dose Point # 1 - (25.1158,16.5,0) in

Energy MeV	Activity Photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0045	7.813e+07	0.000e+00	5.824e-26	0.000e+00	3.992e-26
0.0318	1.558e+08	6.004e-46	9.809e-25	5.001e-48	8.170e-27
0.0322	2.875e+08	3.236e-44	1.839e-24	2.604e-46	1.480e-26
0.0364	1.046e+08	5.466e-32	7.981e-25	3.106e-34	4.535e-27
0.6616	6.772e+09	4.339e+03	1.734e+04	8.412e+00	3.361e+01
Totals	**7.398e+09**	**4.339e+03**	**1.734e+04**	**8.412e+00**	**3.361e+01**

Results - Dose Point # 2 - (1.02e+02,16.5,0) in

Energy MeV	Activity Photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0045	7.813e+07	0.000e+00	3.697e-27	0.000e+00	2.534e-27
0.0318	1.558e+08	1.220e-46	6.226e-26	1.016e-48	5.186e-28
0.0322	2.875e+08	6.465e-45	1.167e-25	5.203e-47	9.394e-28
0.0364	1.046e+08	9.383e-33	5.066e-26	5.331e-35	2.878e-28
0.6616	6.772e+09	2.943e+02	1.139e+03	5.705e-01	2.209e+00
Totals	**7.398e+09**	**2.943e+02**	**1.139e+03**	**5.705e-01**	**2.209e+00**

Results - Dose Point # 3 - (35.366,16.5,0) in

Energy MeV	Activity Photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0045	7.813e+07	0.000e+00	3.008e-26	0.000e+00	2.062e-26
0.0318	1.558e+08	4.320e-46	5.066e-25	3.599e-48	4.220e-27
0.0322	2.875e+08	2.312e-44	9.497e-25	1.860e-46	7.643e-27
0.0364	1.046e+08	3.671e-32	4.122e-25	2.086e-34	2.342e-27
0.6616	6.772e+09	2.400e+03	9.345e+03	4.653e+00	1.812e+01
Totals	**7.398e+09**	**2.400e+03**	**9.345e+03**	**4.653e+00**	**1.812e+01**

Results - Dose Point # 4 - (42.6,16.5,0) in

Energy MeV	Activity Photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0045	7.813e+07	0.000e+00	2.089e-26	0.000e+00	1.432e-26
0.0318	1.558e+08	3.518e-46	3.518e-25	2.931e-48	2.931e-27
0.0322	2.875e+08	1.879e-44	6.596e-25	1.512e-46	5.308e-27
0.0364	1.046e+08	2.939e-32	2.863e-25	1.670e-34	1.627e-27
0.6616	6.772e+09	1.691e+03	6.539e+03	3.277e+00	1.268e+01
Totals	**7.398e+09**	**1.691e+03**	**6.539e+03**	**3.277e+00**	**1.268e+01**

Originator: R.L. Brown Date: July 7, 2004

Checker: R. M Pierson Date: July 7, 2004

Organizational Manager: J. W. Hobbs Date: July 7, 2004

Results - Dose Point # 5 - (115.9,16.5,0) in

Energy MeV	Activity Photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0045	7.813e+07	0.000e+00	2.863e-27	0.000e+00	1.963e-27
0.0318	1.558e+08	1.003e-46	4.822e-26	8.353e-49	4.017e-28
0.0322	2.875e+08	5.307e-45	9.041e-26	4.271e-47	7.276e-28
0.0364	1.046e+08	7.574e-33	3.924e-26	4.303e-35	2.229e-28
0.6616	6.772e+09	2.264e+02	8.784e+02	4.390e-01	1.703e+00
Totals	**7.398e+09**	**2.264e+02**	**8.784e+02**	**4.390e-01**	**1.703e+00**

Results - Dose Point # 6 - (51.65,16.5,0) in

Energy MeV	Activity Photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0045	7.813e+07	0.000e+00	1.429e-26	0.000e+00	9.793e-27
0.0318	1.558e+08	2.832e-46	2.406e-25	2.359e-48	2.004e-27
0.0322	2.875e+08	1.510e-44	4.511e-25	1.216e-46	3.631e-27
0.0364	1.046e+08	2.340e-32	1.958e-25	1.329e-34	1.112e-27
0.6616	6.772e+09	1.161e+03	4.479e+03	2.251e+00	8.683e+00
Totals	**7.398e+09**	**1.161e+03**	**4.479e+03**	**2.251e+00**	**8.683e+00**

Originator: R.L. Brown Date: July 7, 2004

Checker: R. M Pierson Date: July 7, 2004

Organizational Manager: J. W. Hobbs Date: July 7, 2004

Table 2: Output of MicroShield Runs for 7 Drums, Side of TRUPACT-II					
File Name	Point	Dose Rate (mR/hr)	File Name	Point	Dose Rate (mR/hr)
7dp1.0.ms6	1	33.6	7dp1.5.ms6	1	24.2
	2	2.2		2	1.6
	3	18.1		3	13.1
	4	12.7		4	9.1
	5	1.7		5	1.2
	6	8.7		6	6.2
7dp1.1.ms6	1	31.3	7dp1.6.ms6	1	22.9
	2	2.0		2	1.5
	3	16.9		3	12.4
	4	11.8		4	8.6
	5	1.6		5	1.1
	6	8.1		6	5.9
7dp1.2.ms6	1	29.2	7dp1.7.ms6	1	21.7
	2	1.9		2	1.4
	3	15.7		3	11.7
	4	11.0		4	8.2
	5	1.5		5	1.1
	6	7.5		6	5.6
7dp1.3.ms6	1	27.3	7dp1.8.ms6	1	20.6
	2	1.8		2	1.3
	3	14.8		3	11.2
	4	10.3		4	7.8
	5	1.4		5	1.0
	6	7.0		6	5.3
7dp1.4.ms6	1	25.7	7dp1.9.ms6	1	21.1
	2	1.7		2	1.3
	3	13.9		3	11.3
	4	9.7		4	8.2
	5	1.3		5	1.0
	6	6.6		6	5.5
			7dp2.0.ms6	1	18.8
				2	1.2
				3	10.2
				4	7.1
				5	0.9
				6	4.8

Originator: R.L. Brown Date: July 7, 2004

Checker: R. M Pierson Date: July 7, 2004

Organizational Manager: J. W. Hobbs Date: July 7, 2004

MicroShield v6.02 (6.02-00127)
CH2M_Hill_Hanford_Group_Inc.

Page	:1		
DOS File	:drum1.0.ms6	**File Ref**	:Typical Drum Model
Run Date	: May 17, 2004	**Date**	: 5/17/2004
Run Time	: 1:59:33 PM	**By**	: R. L. Brown
Duration	: 00:00:01	**Checked**	: R. M. Pierson

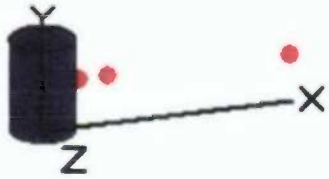
Case Title: TruDrum Side
Description: T-111 Tru Drum Dose Rate Full
Geometry: 7 - Cylinder Volume - Side Shields

Source Dimensions:

Height	84.455 cm	(2 ft 9.3 in)
Radius	28.575 cm	(11.3 in)

Dose Points

A	X	Y	Z
# 1	33.15 cm	41.91 cm	0 cm
	1 ft 1.1 in	1 ft 4.5 in	0.0 in
# 2	228.6 cm	41.91 cm	0 cm
	7 ft 6.0 in	1 ft 4.5 in	0.0 in
# 3	59.182 cm	41.91 cm	0 cm
	1 ft 11.3 in	1 ft 4.5 in	0.0 in



Shields

Shield N	Dimension	Material	Density
Source	1.32e+04 in³	B-202 TRU Custom	1
Transition		Air	0.00122
Air Gap		Air	0.00122
Wall Clad	.053 in	Iron	7.86
Top Clad	.053 in	Iron	7.86

Source Input : Grouping Method - Actual Photon Energies

Nuclide	Curies	becquerels	µCi/cm³	Bq/cm³
Ba-137m	2.0495e-001	7.5830e+009	9.4600e-001	3.5002e+004
Cs-137	2.1664e-001	8.0158e+009	1.0000e+000	3.7000e+004

Buildup : The material reference is - Source
This buildup reference material is a mixed material with a high atomic number element (83). Buildup Factors less than and somewhat greater than 91 keV may be incorrect. Please understand your results.

Integration Parameters

Radial	10
Circumferential	10
Y Direction (axial)	20

Originator: R.L. Brown Date: July 7, 2004

Checker: R. M Pierson Date: July 7, 2004

Organizational Manager: J. W. Hobbs Date: July 7, 2004

Results - Dose Point # 1 - (1.31e+01,16.5,0) in

Energy MeV	Activity Photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0045	7.872e+07	3.249e-36	2.044e-25	2.227e-36	1.401e-25
0.0318	1.570e+08	9.453e-05	1.078e-04	7.874e-07	8.975e-07
0.0322	2.896e+08	2.435e-04	2.783e-04	1.960e-06	2.239e-06
0.0364	1.054e+08	1.642e-03	1.936e-03	9.330e-06	1.100e-05
0.6616	6.823e+09	6.636e+04	1.385e+05	1.286e+02	2.685e+02
Totals	**7.454e+09**	**6.636e+04**	**1.385e+05**	**1.286e+02**	**2.685e+02**

Results - Dose Point # 2 - (90,16.5,0) in

Energy MeV	Activity Photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0045	7.872e+07	7.037e-34	4.777e-27	4.824e-34	3.275e-27
0.0318	1.570e+08	7.779e-06	8.824e-06	6.480e-08	7.350e-08
0.0322	2.896e+08	1.935e-05	2.200e-05	1.557e-07	1.771e-07
0.0364	1.054e+08	9.604e-05	1.127e-04	5.456e-07	6.404e-07
0.6616	6.823e+09	1.632e+03	3.326e+03	3.163e+00	6.447e+00
Totals	**7.454e+09**	**1.632e+03**	**3.326e+03**	**3.163e+00**	**6.447e+00**

Results - Dose Point # 3 - (23.3,16.5,0) in

Energy MeV	Activity Photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0045	7.872e+07	3.924e-33	6.769e-26	2.690e-33	4.639e-26
0.0318	1.570e+08	6.161e-05	6.991e-05	5.132e-07	5.823e-07
0.0322	2.896e+08	1.541e-04	1.753e-04	1.240e-06	1.411e-06
0.0364	1.054e+08	8.345e-04	9.811e-04	4.741e-06	5.574e-06
0.6616	6.823e+09	2.439e+04	4.872e+04	4.728e+01	9.445e+01
Totals	**7.454e+09**	**2.439e+04**	**4.872e+04**	**4.728e+01**	**9.445e+01**

Table 3: Output of MicroShield Run for Single Drum

File Name	Point	Dose Rate (mR/hr)
drum1.0.ms6	1	269
	2	6
	3	94
drum1.1.ms6	1	250
	2	6
	3	89
drum1.2.ms6	1	234
	2	6
	3	83
drum1.3.ms6	1	220
	2	5
	3	78
drum1.4.ms6	1	207
	2	5
	3	74
drum1.5.ms6	1	195
	2	5
	3	70
drum1.6.ms6	1	184
	2	4
	3	67
drum1.7.ms6	1	175
	2	4
	3	63
drum1.8.ms6	1	166
	2	4
	3	60
drum1.9.ms6	1	158
	2	4
	3	58
drum2.0.ms6	1	151
	2	4
	3	55

Originator: R.L. Brown Date: July 7, 2004

Checker: R. M Pierson Date: July 7, 2004

Organizational Manager: J. W. Hobbs Date: July 7, 2004

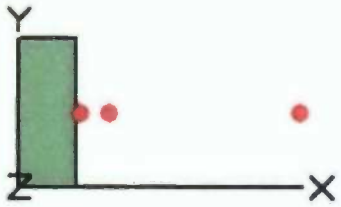
MicroShield v6.02 (6.02-00127)
CH2M_Hill_Hanford_Group_Inc.

Page	:1	**File Ref**	:Typical SWB Bottom Model
DOS File	:swb1.0.ms6	**Date**	: 5/17/2004
Run Date	: May 17, 2004	**By**	: R. L. Brown
Run Time	: 2:14:40 PM	**Checked**	: R. M. Pierson
Duration	: 00:00:02		

Case Title: Standard Waste Box
Description: 20" Full B-202
Geometry: 13 - Rectangular Volume

Source Dimensions:

Length	50.8 cm	(1 ft 8.0 in)
Width	172.72 cm	(5 ft 8.0 in)
Height	132.08 cm	(4 ft 4.0 in)



Dose Points

A	X	Y	Z
# 1	55.58282 cm	66.04 cm	86.36 cm
	1 ft 9.9 in	2 ft 2.0 in	2 ft 10.0 in
# 2	251.46 cm	66.04 cm	86.36 cm
	8 ft 3.0 in	2 ft 2.0 in	2 ft 10.0 in
# 3	81.61782 cm	66.04 cm	86.36 cm
	2 ft 8.1 in	2 ft 2.0 in	2 ft 10.0 in

Shields

Shield N	Dimension	Material	Density
Source	7.07e+04 in³	B-202 TRU Custom	1
Shield 1	.133 in	Iron	7.86
Air Gap		Air	0.00122

Source Input : Grouping Method - Actual Photon Energies

Nuclide	curies	becquerels	µCi/cm³	Bq/cm³
Ba-137m	1.0963e+000	4.0564e+010	9.4600e-001	3.5002e+004
Cs-137	1.1589e+000	4.2879e+010	1.0000e+000	3.7000e+004

Buildup : The material reference is - Source
This buildup reference material is a mixed material with a high atomic number element (83). Buildup Factors less than and somewhat greater than 91 keV may be incorrect. Please understand your results.

Integration Parameters

X Direction	10
Y Direction	20
Z Direction	20

Originator: R.L. Brown Date: July 7, 2004

Checker: R. M Pierson Date: July 7, 2004

Organizational Manager: J. W. Hobbs Date: July 7, 2004

Results - Dose Point # 1 - (21.883,26,34) in

Energy MeV	Activity Photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0045	4.211e+08	3.170e-76	4.609e-25	2.173e-76	3.159e-25
0.0318	8.398e+08	8.482e-10	9.894e-10	7.065e-12	8.241e-12
0.0322	1.549e+09	3.096e-09	3.626e-09	2.492e-11	2.918e-11
0.0364	5.638e+08	4.212e-07	5.155e-07	2.393e-09	2.929e-09
0.6616	3.650e+10	7.357e+04	1.863e+05	1.426e+02	3.612e+02
Totals	**3.987e+10**	**7.357e+04**	**1.863e+05**	**1.426e+02**	**3.612e+02**

Results - Dose Point # 2 - (99,26,34) in

Energy MeV	Activity Photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0045	4.211e+08	2.666e-75	2.467e-26	1.827e-75	1.691e-26
0.0318	8.398e+08	4.667e-10	5.444e-10	3.888e-12	4.534e-12
0.0322	1.549e+09	1.701e-09	1.992e-09	1.369e-11	1.603e-11
0.0364	5.638e+08	2.273e-07	2.781e-07	1.291e-09	1.580e-09
0.6616	3.650e+10	7.286e+03	1.581e+04	1.413e+01	3.065e+01
Totals	**3.987e+10**	**7.286e+03**	**1.581e+04**	**1.413e+01**	**3.065e+01**

Results - Dose Point # 3 - (32.133,26,34) in

Energy MeV	Activity Photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0045	4.211e+08	3.483e-75	2.312e-25	2.388e-75	1.585e-25
0.0318	8.398e+08	6.015e-10	7.017e-10	5.010e-12	5.845e-12
0.0322	1.549e+09	2.212e-09	2.592e-09	1.780e-11	2.086e-11
0.0364	5.638e+08	3.322e-07	4.069e-07	1.887e-09	2.312e-09
0.6616	3.650e+10	5.640e+04	1.292e+05	1.093e+02	2.506e+02
Totals	**3.987e+10**	**5.640e+04**	**1.292e+05**	**1.093e+02**	**2.506e+02**

Table 4: Output of MicroShield Run for Single SWB

File Name	Point	Dose Rate (mR/hr)
swb1.0.ms6	1	361
	2	31
	3	251
swb1.1.ms6	1	331
	2	29
	3	232
swb1.2.ms6	1	305
	2	27
	3	216
swb1.3.ms6	1	283
	2	25
	3	202
swb1.4.ms6	1	263
	2	23
	3	189
swb1.5.ms6	1	246
	2	22
	3	178
swb1.6.ms6	1	231
	2	21
	3	168
swb1.7.ms6	1	218
	2	20
	3	159
swb1.8.ms6	1	206
	2	19
	3	151
swb1.9.ms6	1	195
	2	18
	3	143
swb2.0.ms6	1	185
	2	17
	3	137

Originator: R.L. Brown Date: July 7, 2004

Checker: R. M Pierson Date: July 7, 2004

Organizational Manager: J. W. Hobbs Date: July 7, 2004

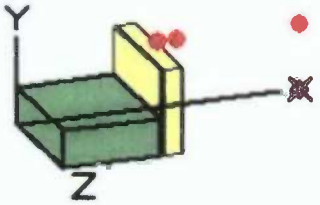
MicroShield v6.02 (6.02-00127)
CH2M_Hill_Hanford_Group_Inc.

Page	:1	**File Ref**	:Typical Model Used for Side of TRUPACT-II Containing Two SWBs
DOS File	:tpboxb1.0.ms6		
Run Date	: May 17, 2004	**Date**	: 5/17/2004
Run Time	: 3:17:30 PM	**By**	: R. L. Brown
Duration	: 00:00:03	**Checked**	: R. M. Pierson

Case Title: Trupact II
Description: 2 each 20" Full SWB in TruPact II From Side
Geometry: 13 - Rectangular Volume

Source Dimensions:

Length	132.08 cm	(4 ft 4.0 in)
Width	175.26 cm	(5 ft 9.0 in)
Height	50.8 cm	(1 ft 8.0 in)



Dose Points

A	X	Y	Z
# 1	165.9128 cm	25.4 cm	76.2 cm
	5 ft 5.3 in	10.0 in	2 ft 6.0 in
# 2	361.045 cm	25.4 cm	76.2 cm
	11 ft 10.1 in	10.0 in	2 ft 6.0 in
# 3	192.024 cm	25.4 cm	76.2 cm
	6 ft 3.6 in	10.0 in	2 ft 6.0 in
# 4	165.9128 cm	116.84 cm	76.2 cm
	5 ft 5.3 in	3 ft 10.0 in	2 ft 6.0 in
# 5	361.045 cm	116.84 cm	76.2 cm
	11 ft 10.1 in	3 ft 10.0 in	2 ft 6.0 in
# 6	192.024 cm	116.84 cm	76.2 cm
	6 ft 3.6 in	3 ft 10.0 in	2 ft 6.0 in

Shields

Shield N	Dimension	Material	Density
Source	7.18e+04 in³	B-202 TRU Custom	1
Shield 1	.133 in	Iron	7.86
Shield 2	1.0 in	Air	0.00122
Shield 3	.25 in	304 SST	7.97
Shield 4	.25 in	Air	0.00122
Shield 5	.188 in	304 SST	7.97
Shield 6	10.125 in	TRUPOLY	0.136
Shield 7	.25 in	304 SST	7.97
Air Gap		Air	0.00122
Immersion		TRUPOLY	0.136

Source Input : Grouping Method - Actual Photon Energies

Nuclide	curies	becquerels	µCi/cm³	Bq/cm³
Ba-137m	1.1124e+000	4.1160e+010	9.4600e-001	3.5002e+004
Cs-137	1.1759e+000	4.3510e+010	1.0000e+000	3.7000e+004

Originator: R.L. Brown Date: July 7, 2004

Checker: R. M Pierson Date: July 7, 2004

Organizational Manager: J. W. Hobbs Date: July 7, 2004

Buildup : The material reference is - Source
This buildup reference material is a mixed material with a high atomic number element (83). Buildup Factors less than and somewhat greater than 91 keV may be incorrect. Please understand your results.

Integration Parameters

X Direction	10
Y Direction	20
Z Direction	20

Results - Dose Point # 1 - (65.32,10,30) in

Energy MeV	Activity Photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0045	4.273e+08	0.000e+00	1.525e-25	0.000e+00	1.045e-25
0.0318	8.521e+08	9.495e-53	2.568e-24	7.909e-55	2.139e-26
0.0322	1.572e+09	8.153e-51	4.815e-24	6.561e-53	3.875e-26
0.0364	5.721e+08	7.871e-37	2.090e-24	4.472e-39	1.187e-26
0.6616	3.704e+10	6.063e+03	2.568e+04	1.175e+01	4.978e+01
Totals	**4.046e+10**	**6.063e+03**	**2.568e+04**	**1.175e+01**	**4.978e+01**

Results - Dose Point # 2 - (1.42e+02,10,30) in

Energy MeV	Activity Photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0045	4.273e+08	0.000e+00	1.587e-26	0.000e+00	1.088e-26
0.0318	8.521e+08	6.937e-53	2.673e-25	5.778e-55	2.227e-27
0.0322	1.572e+09	5.894e-51	5.012e-25	4.743e-53	4.034e-27
0.0364	5.721e+08	5.063e-37	2.175e-25	2.876e-39	1.236e-27
0.6616	3.704e+10	6.213e+02	2.452e+03	1.204e+00	4.753e+00
Totals	**4.046e+10**	**6.213e+02**	**2.452e+03**	**1.204e+00**	**4.753e+00**

Results - Dose Point # 3 - (75.6,10,30) in

Energy MeV	Activity Photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0045	4.273e+08	0.000e+00	9.203e-26	0.000e+00	6.308e-26
0.0318	8.521e+08	9.584e-53	1.550e-24	7.984e-55	1.291e-26
0.0322	1.572e+09	8.214e-51	2.906e-24	6.611e-53	2.339e-26
0.0364	5.721e+08	7.812e-37	1.261e-24	4.439e-39	7.166e-27
0.6616	3.704e+10	3.938e+03	1.613e+04	7.634e+00	3.126e+01
Totals	**4.046e+10**	**3.938e+03**	**1.613e+04**	**7.634e+00**	**3.126e+01**

Results - Dose Point # 4 - (65.32,46,30) in

Energy MeV	Activity Photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0045	4.273e+08	0.000e+00	7.045e-26	0.000e+00	4.829e-26
0.0318	8.521e+08	2.013e-57	1.187e-24	1.677e-59	9.884e-27
0.0322	1.572e+09	2.228e-55	2.225e-24	1.793e-57	1.790e-26
0.0364	5.721e+08	1.999e-40	9.655e-25	1.136e-42	5.486e-27
0.6616	3.704e+10	9.430e+02	5.955e+03	1.828e+00	1.155e+01
Totals	**4.046e+10**	**9.430e+02**	**5.955e+03**	**1.828e+00**	**1.155e+01**

Originator: R.L. Brown Date: July 7, 2004

Checker: R. M Pierson Date: July 7, 2004

Organizational Manager: J. W. Hobbs Date: July 7, 2004

Results - Dose Point # 5 - (1.42e+02,46,30) in

Energy MeV	Activity Photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0045	4.273e+08	0.000e+00	1.440e-26	0.000e+00	9.869e-27
0.0318	8.521e+08	1.698e-53	2.425e-25	1.414e-55	2.020e-27
0.0322	1.572e+09	1.502e-51	4.546e-25	1.209e-53	3.659e-27
0.0364	5.721e+08	1.837e-37	1.973e-25	1.044e-39	1.121e-27
0.6616	3.704e+10	5.680e+02	2.375e+03	1.101e+00	4.604e+00
Totals	**4.046e+10**	**5.680e+02**	**2.375e+03**	**1.101e+00**	**4.604e+00**

Results - Dose Point # 6 - (75.6,46,30) in

Energy MeV	Activity Photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0045	4.273e+08	0.000e+00	5.513e-26	0.000e+00	3.779e-26
0.0318	8.521e+08	2.170e-56	9.285e-25	1.808e-58	7.734e-27
0.0322	1.572e+09	2.257e-54	1.741e-24	1.816e-56	1.401e-26
0.0364	5.721e+08	1.193e-39	7.555e-25	6.775e-42	4.292e-27
0.6616	3.704e+10	1.229e+03	6.722e+03	2.382e+00	1.303e+01
Totals	**4.046e+10**	**1.229e+03**	**6.722e+03**	**2.382e+00**	**1.303e+01**

Table 5: Output of MicroShield Runs for 2 SWBs, Side of TRUPACT-II

File Name	Point	Dose Rate (mR/hr)	File Name	Point	Dose Rate (mR/hr)
tpboxb1.0.ms6	1	50	tpboxb1.6.ms6	1	34
	2	5		2	3
	3	31		3	21
	4	12		4	8
	5	5		5	3
	6	13		6	9
tpboxb1.1.ms6	1	46	tpboxb1.7.ms6	1	32
	2	4		2	3
	3	29		3	20
	4	11		4	7
	5	4		5	3
	6	12		6	8
tpboxb1.2.ms6	1	43	tpboxb1.8.ms6	1	30
	2	4		2	3
	3	27		3	19
	4	10		4	7
	5	4		5	3
	6	11		6	8
tpboxb1.3.ms6	1	40	tpboxb1.9.ms6	1	29
	2	4		2	3
	3	25		3	18
	4	9		4	6
	5	4		5	3
	6	10		6	7
tpboxb1.4.ms6	1	38	tpboxb2.0.ms6	1	28
	2	4		2	3
	3	24		3	17
	4	9		4	6
	5	3		5	2
	6	10		6	7
tpboxb1.5.ms6	1	35			
	2	3			
	3	22			
	4	8			
	5	3			
	6	9			

Originator: R.L. Brown Date: July 7, 2004

Checker: R. M Pierson Date: July 7, 2004

Organizational Manager: J. W. Hobbs Date: July 7, 2004

MicroShield v6.02 (6.02-00127)
CH2M_Hill_Hanford_Group_Inc.

Page	:1	**File Ref**	: Typcial Model for Bottom of TRUPACT II with 2 SWB
DOS File	:tswb1.0.ms6		
Run Date	: May 17, 2004	**Date**	: 5/17/2004
Run Time	: 3:55:56 PM	**By**	: R. L. Brown
Duration	: 00:00:02	**Checked**	: R. M. Pierson

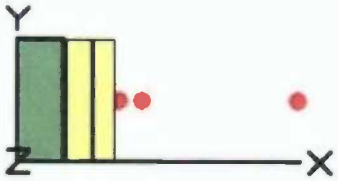
Case Title: Trupact II
Description: 20" Full SWB in TruPact II From Bottom
Geometry: 13 - Rectangular Volume

Source Dimensions:

Length	50.8 cm	(1 ft 8.0 in)
Width	175.26 cm	(5 ft 9.0 in)
Height	132.08 cm	(4 ft 4.0 in)

Dose Points

A	X	Y	Z
# 1	1.10e+02 cm	66.04 cm	86.36 cm
	3 ft 7.2 in	2 ft 2.0 in	2 ft 10.0 in
# 2	1.36e+02 cm	66.04 cm	86.36 cm
	4 ft 5.5 in	2 ft 2.0 in	2 ft 10.0 in
# 3	305 cm	66.04 cm	86.36 cm
	10 ft 0.1 in	2 ft 2.0 in	2 ft 10.0 in



Shields

Shield N	Dimension	Material	Density
Source	7.18e+04 in³	B-202 TRU Custom	1
Shield 1	.135 in	Iron	7.86
Shield 2	.08 in	Aluminum	2.7
Shield 3	11.313 in	Al Hcomb	0.058
Shield 4	.25 in	304 SST	7.97
Shield 5	.25 in	Air	0.00122
Shield 6	.188 in	304 SST	7.97
Shield 7	9.0 in	TRUPOLY	0.136
Shield 8	.25 in	304 SST	7.97
Air Gap		Air	0.00122

Source Input : Grouping Method - Actual Photon Energies

Nuclide	curies	becquerels	µCi/cm³	Bq/cm³
Ba-137m	1.1124e+000	4.1160e+010	9.4600e-001	3.5002e+004
Cs-137	1.1759e+000	4.3510e+010	1.0000e+000	3.7000e+004

Originator: R.L. Brown Date: July 7, 2004

Checker: R. M Pierson Date: July 7, 2004

Organizational Manager: J. W. Hobbs Date: July 7, 2004

Buildup : The material reference is - Source
This buildup reference material is a mixed material with a high atomic number element (83). Buildup Factors less than and somewhat greater than 91 keV may be incorrect. Please understand your results.

Integration Parameters

X Direction	10
Y Direction	20
Z Direction	20

Results - Dose Point # 1 - (43.2155,26,34) in

Energy MeV	Activity Photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0045	4.273e+08	0.000e+00	1.347e-25	0.000e+00	9.233e-26
0.0318	8.521e+08	2.443e-51	2.269e-24	2.035e-53	1.890e-26
0.0322	1.572e+09	1.971e-49	4.253e-24	1.586e-51	3.423e-26
0.0364	5.721e+08	1.112e-35	1.846e-24	6.316e-38	1.049e-26
0.6616	3.704e+10	6.291e+03	2.863e+04	1.220e+01	5.550e+01
Totals	**4.046e+10**	**6.291e+03**	**2.863e+04**	**1.220e+01**	**5.550e+01**

Results - Dose Point # 2 - (53.4655,26,34) in

Energy MeV	Activity Photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0045	4.273e+08	0.000e+00	8.885e-26	0.000e+00	6.090e-26
0.0318	8.521e+08	2.417e-51	1.496e-24	2.014e-53	1.247e-26
0.0322	1.572e+09	1.951e-49	2.806e-24	1.570e-51	2.258e-26
0.0364	5.721e+08	1.103e-35	1.218e-24	6.266e-38	6.918e-27
0.6616	3.704e+10	4.903e+03	2.132e+04	9.505e+00	4.134e+01
Totals	**4.046e+10**	**4.903e+03**	**2.132e+04**	**9.505e+00**	**4.134e+01**

Results - Dose Point # 3 - (1.20e+02,26,34) in

Energy MeV	Activity Photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0045	4.273e+08	0.000e+00	1.669e-26	0.000e+00	1.144e-26
0.0318	8.521e+08	2.258e-51	2.810e-25	1.881e-53	2.341e-27
0.0322	1.572e+09	1.823e-49	5.269e-25	1.467e-51	4.240e-27
0.0364	5.721e+08	1.021e-35	2.287e-25	5.799e-38	1.299e-27
0.6616	3.704e+10	1.116e+03	4.560e+03	2.163e+00	8.840e+00
Totals	**4.046e+10**	**1.116e+03**	**4.560e+03**	**2.163e+00**	**8.840e+00**

Table 6: Output of MicroShield Run for TRUPACT II Bottom with 2 SWB

File Name	Dose Point	Dose Rate (mR/hr
tswb1.0.ms6	1	56
	2	41
	3	9
tswb1.1.ms6	1	51
	2	38
	3	8
tswb1.2.ms6	1	48
	2	36
	3	8
tswb1.3.ms6	1	44
	2	33
	3	7
tswb1.4.ms6	1	42
	2	31
	3	7
tswb1.5.ms6	1	39
	2	29
	3	6
tswb1.6.ms6	1	37
	2	28
	3	6
tswb1.7.ms6	1	35
	2	26
	3	6
tswb1.8.ms6	1	33
	2	25
	3	5
tswb1.9.ms6	1	31
	2	24
	3	5
tswb2.0.ms6	1	30
	2	23
	3	5

Section A-4
Development of Bremsstrahlung Correction Factors

Bremsstrahlung is the process by which an X-Ray is produced by a Beta particle undergoing negative acceleration in the vicinity of a nucleus. The energy of the X-Ray produced is a function of the Z-Number of the atom and the energy of the Beta particle.

The resulting radiation is emitted as a spectrum. This spectrum can be modeled by binning the emitted radiation into energy groupings and modeling the binned energy spectrum as an external source file in MicroShield.

The outcome of this process is shown below as Table 1: Bremsstrahlung Spectrum.

Table 1: Bremsstrahlung Spectrum

Midpoint Energy (MEV)	photon/sec	% of Total Photons	% Energy Contribution
0.015	2.32E+09	29.74%	5.3%
0.025	1.17E+09	15.00%	4.4%
0.035	7.40E+08	9.49%	3.9%
0.045	5.23E+08	6.70%	3.6%
0.055	3.95E+08	5.06%	3.3%
0.07	5.61E+08	7.19%	6.0%
0.09	3.84E+08	4.92%	4.8%
0.11	2.80E+08	3.59%	4.7%
0.14	3.82E+08	4.90%	8.1%
0.18	2.47E+08	3.16%	6.8%
0.225	2.06E+08	2.64%	7.0%
0.285	1.84E+08	2.36%	8.0%
0.36	1.30E+08	1.67%	7.1%
0.45	9.96E+07	1.28%	6.8%
0.565	7.52E+07	0.96%	6.5%
0.71	5.00E+07	0.64%	5.4%
0.895	3.18E+07	0.41%	4.3%
1.13	1.59E+07	0.20%	2.7%
1.4	5.55E+06	0.07%	1.2%
1.69	1.35E+06	0.02%	0.0%
1.95	1.27E+05	0.00%	0.0%
2.175	6.22E+03	0.00%	0.0%

Table 1 is the spectrum of Bremsstrahlung photons emitted by 1 Curie of Strontium 90 in equilibrium with its daughter product Yttrium 90. The ratio of the calculated dose rates

for a Curie of Sr:90 to a Curie of Cs137 was found to be 0.013:1. This resulted in the development of correction factors for dose rates based on the Sr:Cs ratios. A 10:1 Sr:Cs ratio will provide a calculated dose rate that is approximately 1.13 times higher than the Cs dose rate for the materials being packaged as waste.

This led to the development of Table 2, which shows the Bremsstrahlung factor for each tank. By multiplying the Cs-137 concentration in the tank or the dose rate from Cesium by the Bremsstrahlung Correction Factor, the dose rate can be corrected to account for Strontium in the waste.

Table 2: Bremsstrahlung Correction Factors for Tanks

Tank	Cesium (Ci)	Strontium (Ci)	Ratio (Sr/Cs)	Bremsstrahlung Correction Factor
241-B-201	2.11E+01	2.67E+02	13:1	1.2
241-B-202	1.11E+01	4.37E+02	39:1	1.5
241-B-203	1.92E+00	1.73E+01	9:1	1.1
241-B-204	6.67E+00	3.15E+02	47:1	1.6
241-T-104	2.57E+02	3.38E+03	13:1	1.2
241-T-110	2.78E+01	5.12E+01	2:1	1.0
241-T-111	2.29E+02	9.23E+03	40:1	1.5
241-T-201	4.40E+00	1.13E+01	3:1	1.0
241-T-202	2.61E+00	2.30E-01	0.1:1	1.0
241-T-203	2.51E+00	4.07E-01	0.2:1	1.0
241-T-204	1.24E+00	7.32E-01	0.6:1	1.0

U.S. Department of Energy
Office of River Protection
P.O. Box 450
Richland, Washington 99352

0059513

MAY 15 2003

03-ED-061

Mr. Michael A. Wilson, Program Manager
Nuclear Waste Program
State of Washington
Department of Ecology
1315 W. Fourth Avenue
Kennewick, Washington 99336

RECEIVED
MAY 29 2003
EDMC

Dear Mr. Wilson:

NOTICE OF INTENT (NOI) TO SUBMIT A PERMIT APPLICATION FOR PROPOSED CONTACT-HANDLED TRANSURANIC MIXED WASTE PACKAGING AND INTERIM STORAGE FACILITY

Attached is the U.S. Department of Energy, Office of River Protection, NOI to file a permit application for a Contact-Handled Transuranic Mixed Waste Packaging and Interim Storage Facility Resource Conservation and Recovery Act Part B Application. Per Washington Administrative Code 173-303-281, this NOI must be filed with the Washington State Department of Ecology no less than 150 days prior to filing the certified Part B Permit Application (Revision 0).

As an on-going activity, we have been meeting with members of your staff on a bi-weekly basis through the Mission Acceleration Initiative for the Supplemental Technologies Project. At our meeting on April 8, 2003, we informed them that we were preparing the NOI for submission. We also agreed to work on our process to expedite preparation, submittal and review of the permit application.

If you have any questions, please contact Richard R. McNulty, Environmental Division, (509) 373-9304, or Robert M. Yasek, Tank Farms Programs and Projects Division, (509) 372-1270.

Sincerely,

James E. Rasmussen

James E. Rasmussen, Director
Environmental Division

ED:LAH

Attachment

cc: See page 2

cc w/o attach:
B. G. Erlandson, BNI
D. I. Allen, CHG
W. T. Dixon, CHG
M. Jarayssi, CHG
J. Kristofzski, CHG
F. Miera, CHG
S. L. Dahl, Ecology
N. Ceto, EPA
J. L. Hanson, INNOV
J. B. Hebdon, RL
T. C. McKarns, RL
Administrative Record
Environmental Portal, LMSI

Attachment
03-ED-061

Notice of Intent for Proposed Operation of Contact-Handled Transuranic Mixed Waste Packaging Subsystems

NOTICE OF INTENT FOR PROPOSED CONTACT-HANDLED TRANSURANIC MIXED WASTE PACKAGING AND INTERIM STORAGE FACILITY

MAY 2003

U.S. Department of Energy
Office of River Protection
Richland, Washington

Prepared by
CH2M HILL Hanford Group, Inc.
Richland, Washington

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ACRONYMS

CH-TRUM	contact-handled transuranic mixed waste
DOE	U.S. Department of Energy
Ecology	State of Washington, Department of Ecology
NOI	notice of intent
ORP	U.S. Department of Energy, Office of River Protection
Part B	dangerous waste permit application
PNNL	Pacific Northwest National Laboratory
RCRA	*Resource Conservation and Recovery Act of 1976*
SEPA	Washington State Environmental Policy Act
TSD	treatment, storage, and/or disposal
WAC	*Washington Administrative Code*
WIPP	Waste Isolation Pilot Plant

METRIC CONVERSION CHART

From common U.S. units into metric			From metric units into common U.S.		
If you know	Multiply by	To get	If you know	Multiply by	To get
Length			**Length**		
inches	25.40	millimeters	millimeters	0.0393	inches
inches	2.54	centimeters	centimeters	0.393	inches
feet	0.3048	Meters	Meters	3.2808	feet
yards	0.914	Meters	Meters	1.09	yards
miles	1.609	Kilometers	kilometers	0.62	miles
Area			**Area**		
square inches	6.4516	square centimeters	square centimeters	0.155	square inches
square feet	0.092	square meters	square meters	10.7639	square feet
square yards	0.836	square meters	square meters	1.20	square yards
square miles	2.59	square kilometers	square kilometers	0.39	square miles
acres	0.404	Hectares	Hectares	2.471	acres
Mass (weight)			**Mass (weight)**		
ounces	28.35	Grams	Grams	0.0352	ounces
pounds	0.453	Kilograms	Kilograms	2.2046	pounds
short ton	0.907	metric ton	metric ton	1.10	short ton
Volume			**Volume**		
fluid ounces	29.57	Milliliters	Milliliters	0.03	fluid ounces
quarts	0.95	Liters	Liters	1.057	quarts
gallons	3.79	Liters	Liters	0.26	gallons
cubic feet	0.03	cubic meters	cubic meters	35.3147	cubic feet
cubic yards	0.76456	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	0.948	British thermal unit per second	British thermal unit per second	1.055	kilowatt
Force/Pressure			**Force/Pressure**		
pounds per square inch	6.895	Kilopascals	kilopascals	0.14504	pounds per square inch

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

1.0 INTRODUCTION

The first step in obtaining a permit under the State of Washington Department of Ecology (Ecology) *Dangerous Waste Regulations*, Washington Administrative Code (WAC) 173-303-281, calls for dangerous waste facility owners and/or operators to submit a Notice of Intent (NOI). Submittal of a Part A permit application, Form 3, and/or dangerous waste permit application (Part B) for new or expanded dangerous waste treatment, storage, and/or disposal (TSD) units may follow the NOI.

This document serves as notice for the proposed construction and operation on the Hanford Site for a Waste Packaging and Interim Storage Facility (to consist of two identical subsystems) to process and store tank contact-handled transuranic mixed (CH-TRUM) waste. (Transuranic mixed waste means waste containing more than 100 nanocuries of alpha-emitting isotopes per gram of waste, with half-lives greater than 20 years with a hazardous waste component). In addition to these activities, the facility will be used to prepare and certify the CH-TRUM for shipment to the U.S. Department of Energy (DOE) Waste Isolation Pilot Plant (WIPP) facility for disposal. This portion of the activity (e.g., visual examination) is a dual use activity which does not require a permit under the WAC, and will not be further described for the purposes of the permit NOI. Each CH-TRUM storage and processing facility will consist of an interim storage area and three tanks plus a dewatering device to meet interim storage requirements (miscellaneous unit) designed in accordance with WAC 173-303.

The proposed construction and operation of the waste packaging and interim storage facility is being pursued in accordance with three sets of criteria: (1) WAC 173-303; (2) the *Resource Conservation and Recovery Act of 1976* (RCRA), as amended, and 40 CFR 260.1 et seq.; and (3) the waste acceptance criteria for disposal at WIPP.

The following identifies the operator of the CH-TRUM waste packaging and interim storage facility and the contact:

Owner/Operator:	U.S. Department of Energy, Office of River Protection Manager, Office of River Protection: Mr. Roy J. Schepens
Owner:	Hanford Site Wide Permit U.S. Department of Energy Richland Operations Office Manager, Keith A. Klein
Contact:	Mr. Roy J. Schepens, Manager, Office of River Protection
Address:	U.S. Department of Energy, Office of River Protection Post Office Box 450 Richland, Washington 99352 Telephone: (509) 376-6677

As is stated in the Hanford Federal Facility Agreement and Consent Order between DOE, Ecology, and the U.S. Environmental Protection Agency (EPA), DOE agrees that RCRA TSD units are subject to the regulatory framework of Chapter 70.105 RCW pursuant to RCRA Section 6001. Agreement, Article I, Paragraph 6. As is also stated in the Tri-Party Agreement, however, "nothing in the Agreement shall be construed to require DOE to take any action pursuant to RCRA which is inconsistent with the requirements of the Atomic Energy Act of 1954, as amended." Agreement, Article I, Paragraph 5. Accordingly, the Owner/Operator submits this NOI only with respect to the materials specifically described herein. The Owner/Operator does not intend the NOI to imply, and therefore hereby reserves the right to deny, that application of Chapter 70.105 RCW to the materials and activities described herein extends to other materials or other activities and further reserves the right to assert the Land Disposal Restriction exemptions of the Land Withdrawal Act as appropriate.

2.0 FACILITY DESCRIPTION AND GENERAL PROVISIONS

The Hanford Site is a single RCRA facility identified by the EPA/State Identification Number WA7890008967 that consists of 70 TSD units that have or are conducting dangerous waste management activities. These TSD units are included in the Hanford Facility Dangerous Waste Permit (DOE/RL-88-21). The Tank CH-TRUM Waste Packaging and Interim Storage Facility will be included in the Hanford Facility Site Wide RCRA Permit. This is an expansion of the existing permit.

The following sections provide a description of the Tank CH-TRUM Waste Packaging and Interim Storage Facility, support equipment systems, and other general provisions specified in WAC 173-303-281.

2.1 LOCATION OF PROPOSED EXPANSION

Two CH-TRUM Waste Packaging and Interim Storage Subsystems will be constructed. One will be located in the 200 East Area in the 241-B tank farm to support retrieval efforts from the B-200 series tanks. The second will be located in the 200 West Area in the 241-T tank farm to support retrieval efforts from the T-200 series tanks. The subsystems then will be relocated in order to simultaneously package waste from Tank 241-T-111. All sites are located on the Hanford Site, in Benton County, Washington.

Large-scale maps and a general site plan are provided in Appendix A and include the following:

- General overview of the Hanford Site (H-6-958), Figure A-1.
- Site plans showing the 241-B and 241-T tank farm locations, Figures A-2 and A-3.
- Site plans showing the layout of the Tank CH-TRUM Waste Packaging Subsystems at the 200 East and 200 West Areas, Figures A-4, A-5, and A-6. Topographic maps

consistent with the criteria of WAC 173-303-806(4)(a)(xviii) will be provided in the Part B Permit Application.

2.2 DESCRIPTION OF PROPOSED FACILITY

The primary objective for the Tank CH-TRUM Waste Packaging Subsystems is to package, for interim storage prior to shipment to WIPP, CH-TRUM wastes into containers that are compliant for storage. The total layout aerial extent for the CH-TRUM Waste Packaging Subsystem at the 241-B and the 241-T tank farm areas is estimated at approximately 9,600 sq ft, not including the interim storage area for which the size is yet to be determined. As documented by RPP-13300, and RPP-13873, the wastes in the tanks are CH-TRUM containing waste codes as listed in the *Single-Shell Tank System Part A Application*, Revision 6 (DOE/RL-88-21). The total estimated quantity of waste to be processed by these systems is 731,000 gal (2,800 m^3). Based on an assumed throughput processing packaging volume of 2,880 gallons per day (11 m^3/day), we anticipate this total volume will require approximately 15 to 18 months to process.

The tank waste streams will be retrieved and sent to the associated Waste Packaging Facility (see Figure A-6). A slurry will be developed by adding process water to the CH-TRUM sludge waste, as it is retrieved from the tanks. This slurry will be collected by the waste retrieval system and conveyed to the Waste Packaging Subsystems. The subsystems will route the wastes through a centrifuge or other dewatering device, as required. The dewatered solid stream then will be mixed with absorbent (to be identified) to prevent free liquid formation during transportation. The solid sludge-absorbent mixture will be packaged in either 55-gallon drums or solid waste boxes. Once packaged, the CH-TRUM solid waste will be interim stored in this facility and subsequently moved for shipment, or to another permitted storage facility onsite, for example the 200 Area Central Waste Complex, prior to shipment to WIPP.

Liquid effluent resulting from the separations process initially will be reused for the retrieval and slurry of wastes. After completion of the retrieval efforts, remaining liquid effluent will be characterized and transported via container truck to either the permitted double-shell tank system or the permitted Effluent Treatment Facility.

Secondary containment will be integrated into components of the Tank CH-TRUM Waste Packaging Subsystem. Systems will be constructed to minimize the impacts to human health and the environment and to be consistent with the criteria of WAC 173-303.

The following is a list of major equipment items proposed for each of the two waste packaging subsystems. Note that dimensions, sizing, and/or quantities of the subsystem components will be defined as the system designs mature:

- Full container sealing, decontamination, and survey area with separate load-out storage area prior to transport to storage facility;
- Receiver tanks;

- Dewatering device;
- Processed waste packaging containers;
- Liquid waste holding tank with pumps and filters;
- Air emission control devices;
- Water tanker for removal of excess process water;
- Absorbent dispenser/sorbent materials with mixers for waste/absorbent in containers;
- Empty container holding and preparation area(s);
- Change facility/office space; and
- System control area.

2.3 DESCRIPTION OF NEED FOR FACILITY

The proposed transportable packaging units will be unique to the state of Washington for handling transuranic mixed tank wastes. The proposed units are required to provide a capability to package CH-TRUM waste for interim storage, and in a manner consistent with, and that will meet the requirements for long term disposal of this waste in DOE WIPP deep geological repository in New Mexico. Other facilities on the Hanford Site do not have the capability to process tank waste in the required packaging configuration.

2.4 COMPLIANCE WITH STATE ENVIRONMENTAL POLICY ACT

A *State Environmental Policy Act* checklist will be provided with the submittal of the dangerous waste permit application. In addition, the Tank CH-TRUM Waste Packaging Facility project will comply with the environmental requirements of the *National Environmental Policy Act of 1969*.

2.5 COMPLIANCE WITH SITING STANDARDS

Demonstration of compliance with the siting criteria as required under WAC 173-303-282(6) and (7) is addressed in the following sections.

2.5.1 Criteria for Elements of the Natural Environment

The following addresses measures that will be in place for the Tank CH-TRUM Waste Packaging Subsystems to protect the natural environment. Each element of the criteria identified in WAC 173-303-282(6) is addressed.

2.5.1.1 **Earth.** This section addresses the potential for the release of waste into the environment because of structural damage resulting from conditions of the earth at the Tank CH-TRUM Waste Packaging Subsystems.

2.5.1.1.1 Seismic Consideration. No active faults, or evidence of a fault that has had displacement during Holocene times, have been found at the Hanford Site (DOE/RL-0164). The youngest faults recognized on the Hanford Site occur on Gable Mountain, over 4.5 kilometers north of the 200 East Area. These faults are of Quaternary age and are considered "capable" by the Nuclear Regulatory Commission (NUREG-0892).

The Tank CH-TRUM Waste Packaging Subsystems at both the 241-B and 241-T tank farms will be designed and located in Zone 2B as identified in the Uniform Building Code (ICBO 1991). The design of the Tank CH-TRUM Waste Packaging Subsystems for seismic risk will be evaluated in accordance with DOE 6430.1A, *General Design Criteria.*

2.5.1.1.2 Subsidence. The Tank CH-TRUM Waste Packaging Subsystems will be located in the 200 East and 200 West Areas of the Hanford Site. These areas of the Hanford Site are not considered subject to subsidence (PNNL-6415).

2.5.1.1.3 Slope or Soil Instability. The Tank CH-TRUM Waste Packaging Subsystems will not be located in an area of slope or soil instability, or in an area affected by unstable slope or soil conditions (PNNL-6415).

2.5.1.2 **Air.** The Tank CH-TRUM Waste Packaging Units will not utilize incineration, but will have processing tanks and packaging systems tied to an emissions control system. Air emission requirements will be identified as the system design matures and as part of the Part B permitting process. Discussion of measures taken to reduce air emissions resulting from incineration is not applicable.

2.5.1.3 **Water.** This section addresses the potential for contaminating water of the State in the event of a release of waste.

2.5.1.3.1 Surface Water. The following sections address considerations for the protection of surface water.

Flood, Seiche, and Tsunami Protection

Three sources of potential flooding of the Hanford Site are considered: (1) the Columbia River, (2) the Yakima River, and (3) storm-induced run-off in ephemeral streams draining the Hanford Site. No perennial streams occur in the central part of the Hanford Site. Figure 2 shows the 100-year flood plain of the Columbia River, Yakima River, and the Cold Creek probable maximum flood.

The flow of the Columbia River is controlled largely by several upstream dams that are designed to reduce major flood flows. The probable maximum flood for the Columbia River downstream of Priest Rapids Dam has been calculated to be 40,000 cubic meters per second (m^3/s). The flow is greater than the 500-year flood, and although this flood would inundate parts of the 100 Area located adjacent to the Columbia River, this flood would not impact the central plateau on the Hanford Site (i.e., 200 East or 200 West Areas) (PNNL-6415).

The maximum flood recorded in the Yakima River at Kiona, Washington, was 1,900 m^3/s during December 1933. The recurrence interval for the 1933 flood is estimated to be 170 years. The flood only impacted the southernmost part of the Hanford Site in the vicinity of the Horn Rapids Dam. Since that flood there have been significant impoundments in the Yakima River Basin to support irrigation that reduces this threat. The overall magnitude of the flow between the Columbia River and Yakima River (40,000 m^3/s versus 1,900 m^3/s) renders the threat of flooding from the Yakima River to be insignificant in comparison to that from the Columbia River (PNNL-6415).

The only other potential source of flooding of the Hanford Site is run-off from a large precipitation event in the Cold Creek watershed. This event could result in flooding of the ephemeral Cold Creek. Pacific Northwest National Laboratory (PNL-4219) has given an estimate of the probable maximum flood using conservative values of precipitation, infiltration, surface roughness, and topographic features. The impact associated with the maximum flood in the Cold Creek watershed would be limited to portions of land along State Route 240 (PNNL-6415).

The results from past hydrologic analysis associated with the potential flooding of the Columbia River, Yakima River, and Cold Creek watershed show that these waters would not impact the proposed Tank CH-TRUM Waste Packaging Subsystems.

The Hanford Site is not located in an area subject to seiches or coastal flooding, including tsunami or storm surges.

Perennial Surface Water Bodies

The Tank CH-TRUM Waste Packaging Subsystems are both non-land-based facilities as defined in WAC 173-303-282(3)(i). The nearest perennial surface water body is well in excess of the 500 feet from the systems as required by WAC 173-303-282(6)(c)(i)(B)(I).

Surface Water Supply

The Tank CH-TRUM Waste Packaging Subsystems will not be located within any area designated as a watershed and will be located well in excess of 500 feet from the nearest surface water intake for domestic water, consistent with WAC 173-303-82(6)(c)(i)(C)(II).

2.5.1.3.2 Groundwater. The following sections address consideration for the protection of groundwater. The Tank CH-TRUM Waste Packaging Subsystems will be non-land-based units as defined by WAC 173-303-282(3)(i). These facilities will be located outside of both the 100-year and 500-year flood plains.

Depth to Groundwater

The depth to groundwater in both the 200 East and 200 West Areas generally is greater than 60 meters.

Groundwater Management Areas and Special Protection Areas

The proposed construction and operation of the two waste packaging subsystems will minimize groundwater impacts through the use of secondary containment systems and is not expected to result in an increased potential for release of mixed waste to the groundwater or to a special protection area.

Groundwater Intakes

The Tank CH-TRUM Waste Packaging Subsystems will not be located within 500 feet of any groundwater intake for domestic water, consistent with the criteria in WAC 173-303-282(6)(c)(ii)(D)(I).

2.5.1.4 **Plants and Animals**. The following sections address consideration to reduce the potential for waste contaminating plant and animal habitat in the event of a release of waste. The Tank CH-TRUM Waste Packaging Subsystems will be over 0.4 kilometers from any of the following.

2.5.1.4.1 **Wetlands**. The Tank CH-TRUM Waste Packaging Subsystems will not be located near any wetlands.

2.5.1.4.2 **Designated Critical Habitat**. The Tank CH-TRUM Waste Packaging Subsystems will not be located in areas designated as critical habitat for federally listed threatened or endangered species as defined by the *Endangered Species Act of 1973*.

2.5.1.4.3 **State Designated Habitat**. The Tank CH-TRUM Waste Packaging Subsystems will not be located in areas designated by the Washington State Department of Wildlife as habitat essential to the maintenance or recovery of any state-listed threatened or endangered species.

2.5.1.4.4 **Natural Area Preserves**. The Tank CH-TRUM Waste Packaging Subsystems will not be located in any natural area acquired or voluntarily registered or dedicated under Chapter 79.70 Revised Code of Washington.

2.5.1.4.5 **Wildlife Refuge, Preserve, or Bald Eagle Protection Area**. The Tank CH-TRUM Waste Packaging Subsystems will not be located in a state or federally designated wildlife refuge, preserve, or bald eagle protection area.

2.5.2 **Criteria for Elements of the Built Environment**

The following sections address the locational factors affecting protection of the built environment. Each element of the criteria for non-land-based facilities or units identified in WAC 173-303-282(7) is addressed.

2.5.2.1 **Adjacent Land Use.** This section addresses the setback criteria for adjacent land use. The Tank CHTRUM Waste Packaging Subsystems in the 200 East Area will be located approximately 400 meters from the boundary for the 200 East Area. Similarly, the Tank

CH-TRUM Waste Packaging and Support Equipment System in the 200 West Area will be located approximately 900 meters from the boundary of the 200 West Area.

2.5.2.2 **Special Land Uses.** This section addresses setback criteria for special land uses.

2.5.2.2.1 Wild and Scenic Rivers. The Tank CH-TRUM Waste Packaging Subsystems located in the 200 East Area will be at least 11 kilometers from the Columbia River, which has been proposed as a Wild and Scenic River. Similarly, the Tank CH-TRUM Waste Packaging Subsystems located in the 200 West Area will be at least 9 kilometers from the Columbia River. The Tank CH-TRUM Waste Packaging Subsystems will not be within the viewshed of those using the Columbia River.

2.5.2.2.2 Parks, Recreation Areas, National Monuments. The Tank CH-TRUM Waste Packaging Subsystems will be situated significantly farther than 0.4 kilometers from the nearest state or federally designated park, recreation area, or national monument. The Tank CH-TRUM Waste Packaging Subsystems are not within 0.4 kilometers of the Hanford Reach National Monument (65 FR 37253).

2.5.2.2.3 Wilderness Area. The Tank CH-TRUM Waste Packaging Subsystems will be located more than 0.4 kilometers from any wilderness areas as defined by the *Wilderness Act of 1964*.

2.5.2.2.4 Farmland. The Tank CH-TRUM Waste Packaging Subsystems will be more than 0.4 kilometers from any commercial or private prime farmland.

2.5.2.3 **Residences and Public Gathering Places.** This section discusses factors affecting residences and public gathering places. The Tank CH-TRUM Waste Packaging Subsystems will be located more than 0.4 kilometers from residences and public gathering places.

2.5.2.3.1 Incineration. Incineration will not be a process used at the Tank CH-TRUM Waste Packaging Subsystem.

2.5.2.3.2 Land Use Compatibility. The Hanford Site conforms to local land use zoning designation requirements.

2.5.2.3.3 Archeological Sites and Historic Sites. There are no known archaeological or Native American religious sites on or next to the proposed Tank CH-TRUM Waste Packaging Subsystems sites.

3.0 TEN-YEAR COMPLIANCE HISTORY

Appendix B contains formal notices of Hanford Site violations and/or notices of penalties of the Applicant, the U.S. Department of Energy, in accordance with WAC 173-303-281.

4.0 JUSTIFICATION OF NEED

The U.S. Department of Energy issued an Integrated Mission Acceleration Plan (RPP-13678) specifically identifying the accelerated removal of CH-TRUM wastes to accelerate the closure of Hanford Site tank farm single-shell tanks. Sixty-seven of the 149 single-shell tanks have been reported to have leaked. The proposed facility, described herein, is key to the removal of CH-TRUM waste from Hanford Site.

5.0 IMPACT ON OVERALL CAPACITY ON THE HANFORD SITE AND THE STATE OF WASHINGTON

The current capacity for the processing of CH-TRUM waste is limited within Washington State and on the Hanford Site. The CH-TRUM Waste Packaging Facility will provide the means for responsible storage of Hanford Tank Farms CH-TRUM wastes and will comply with all relevant and applicable criteria. The packaged CH-TRUM waste will be disposed in the DOE deep geologic repository Waste Isolation Pilot Plant in New Mexico. This proposed facility will support the current on-site mission of waste management and environmental restoration and remediation.

6.0 REFERENCES

65 FR 37253, 2000, Proclamation 7319 of June 9, 2000, "Establishment of the Hanford Reach National Monument," *Federal Register*, June 13, 2000.

COE, 1969, *Lower Columbia River Standard Project Flood and Probable Maximum Flood*, U.S. Army Corps of Engineers, North Pacific Division, Portland, Oregon.

DOE/EIS-0189-SA2, 1998, *Supplement Analysis for the Tank Waste Remediation System*, U.S. Department of Energy, Washington, D.C.

DOE/EIS-0189-SA3, 2000, *Supplement Analysis for the Tank Waste Remediation System*, U.S. Department of Energy, Washington, D.C.

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DOE Order 6430.1A, *General Design Criteria*, U.S. Department of Energy, Washington D.C.

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RPP-13678, 2003, *Integrated Mission Acceleration Plan*, Rev. 0, CH2M HILL, Richland, Washington.

RPP-13873, 2003, *Origin and Classification of Wastes in Single-Shell Tanks 241-T-110 and 241-T-111*, CH2M HILL, Richland, Washington.

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Figure 1. Hanford Site.

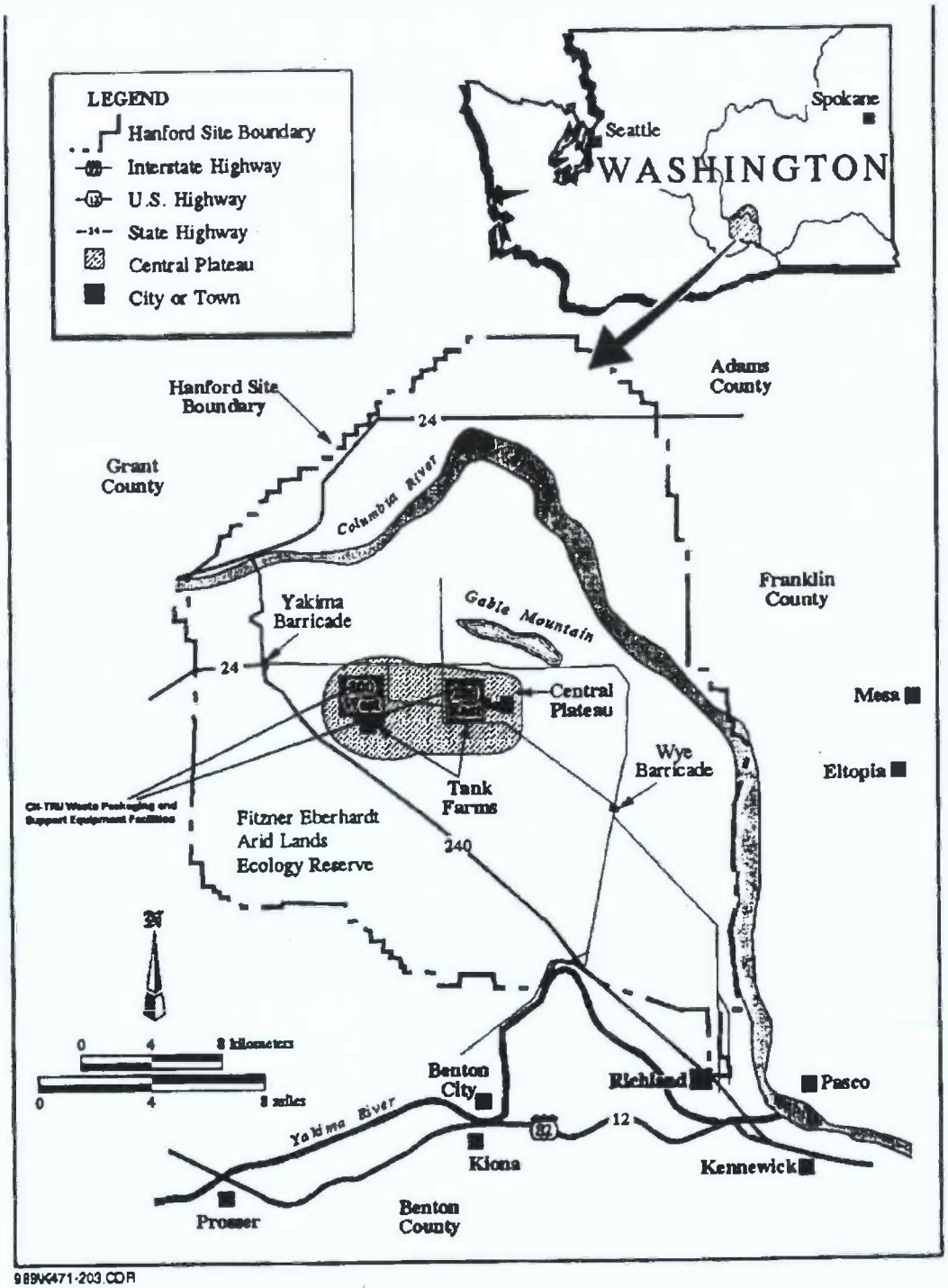
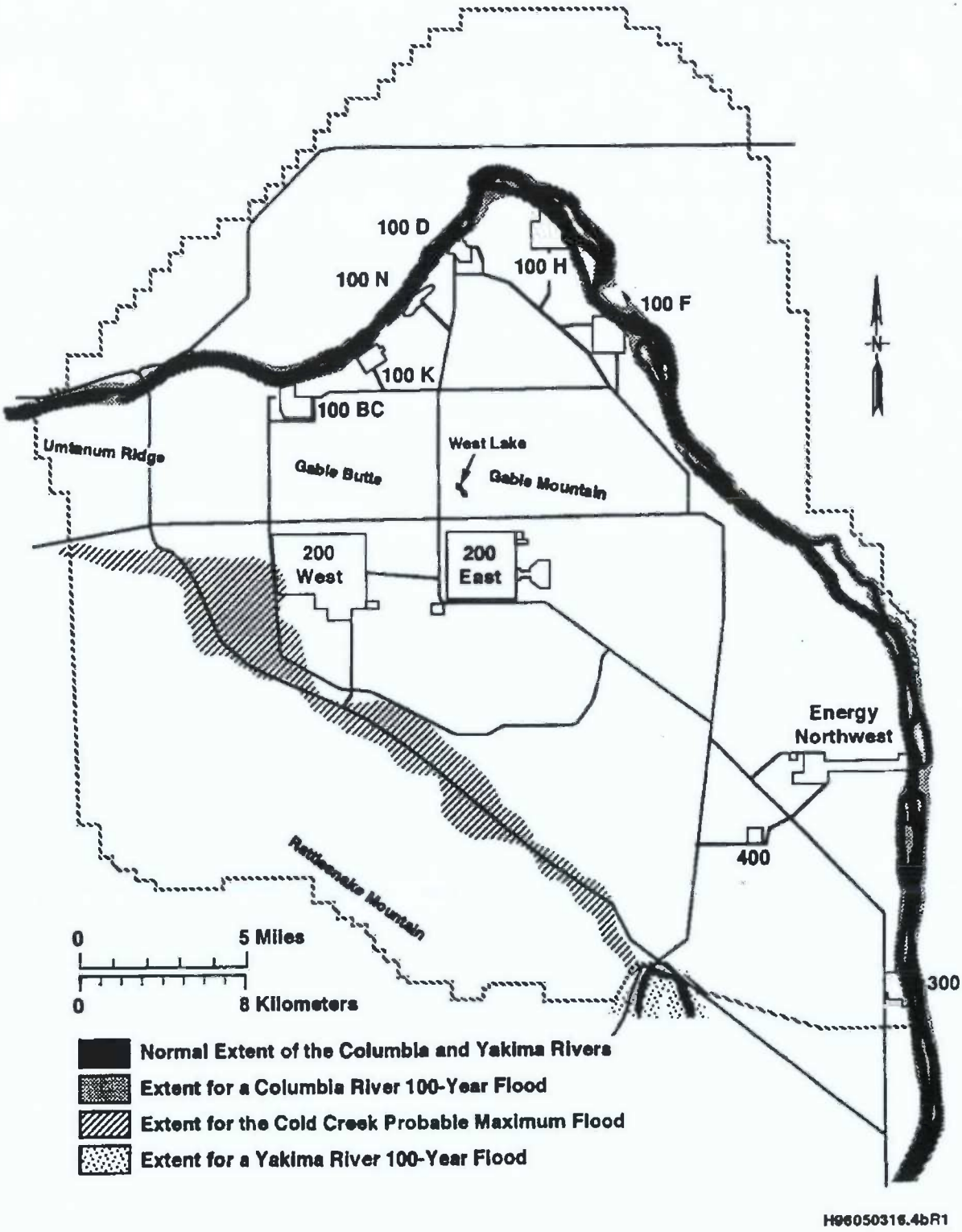


Figure 2. 100-Year Floodplain of the Columbia River and Yakima River and the Cold Creek Probable Maximum Flood.



APPENDIX A

LOCATION MAPS

Figure A-1. General Overview of Hanford Site (H-6-958).

- Official Use Only -

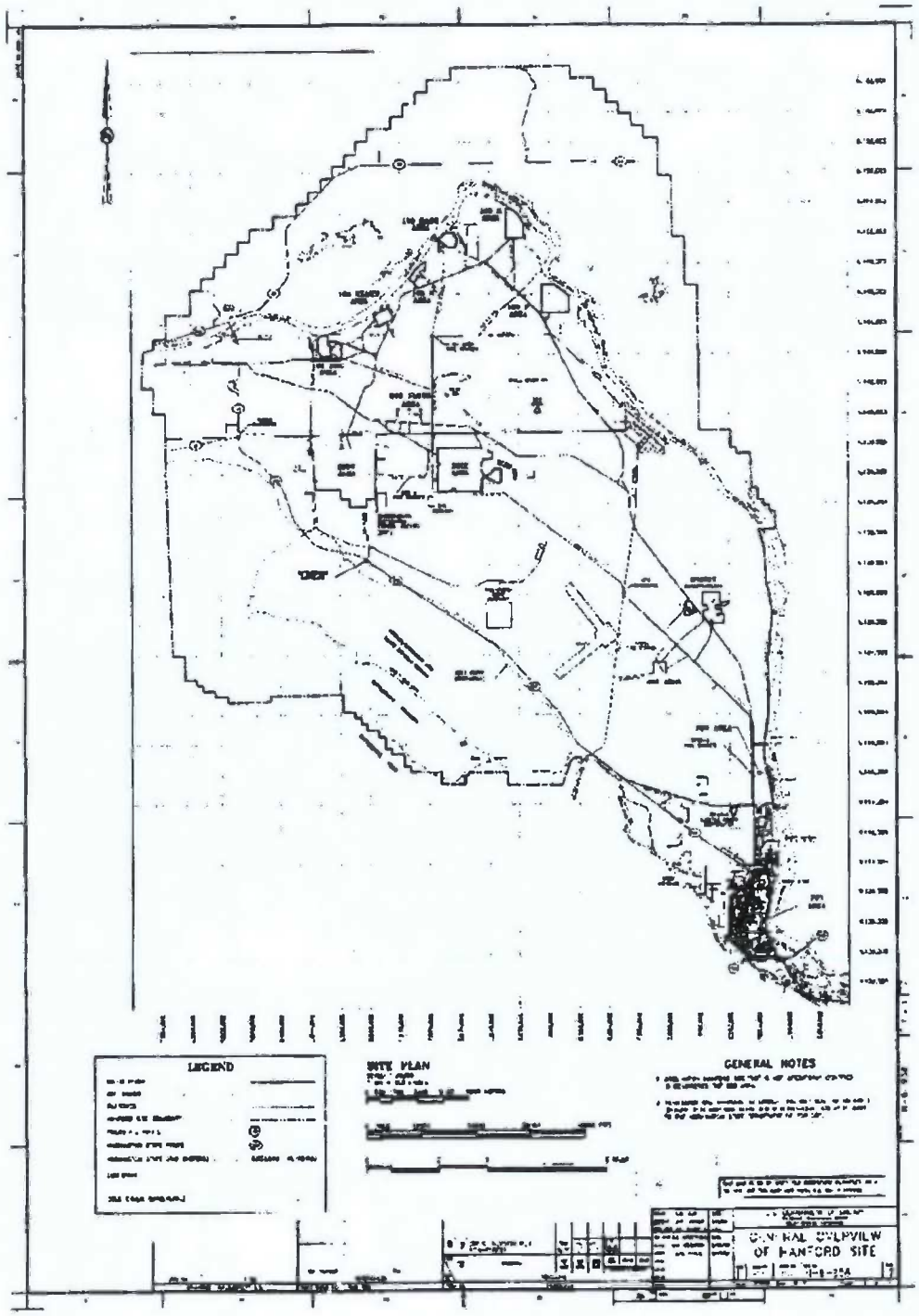


Figure A-2. 200 East Area.
- Official Use Only -

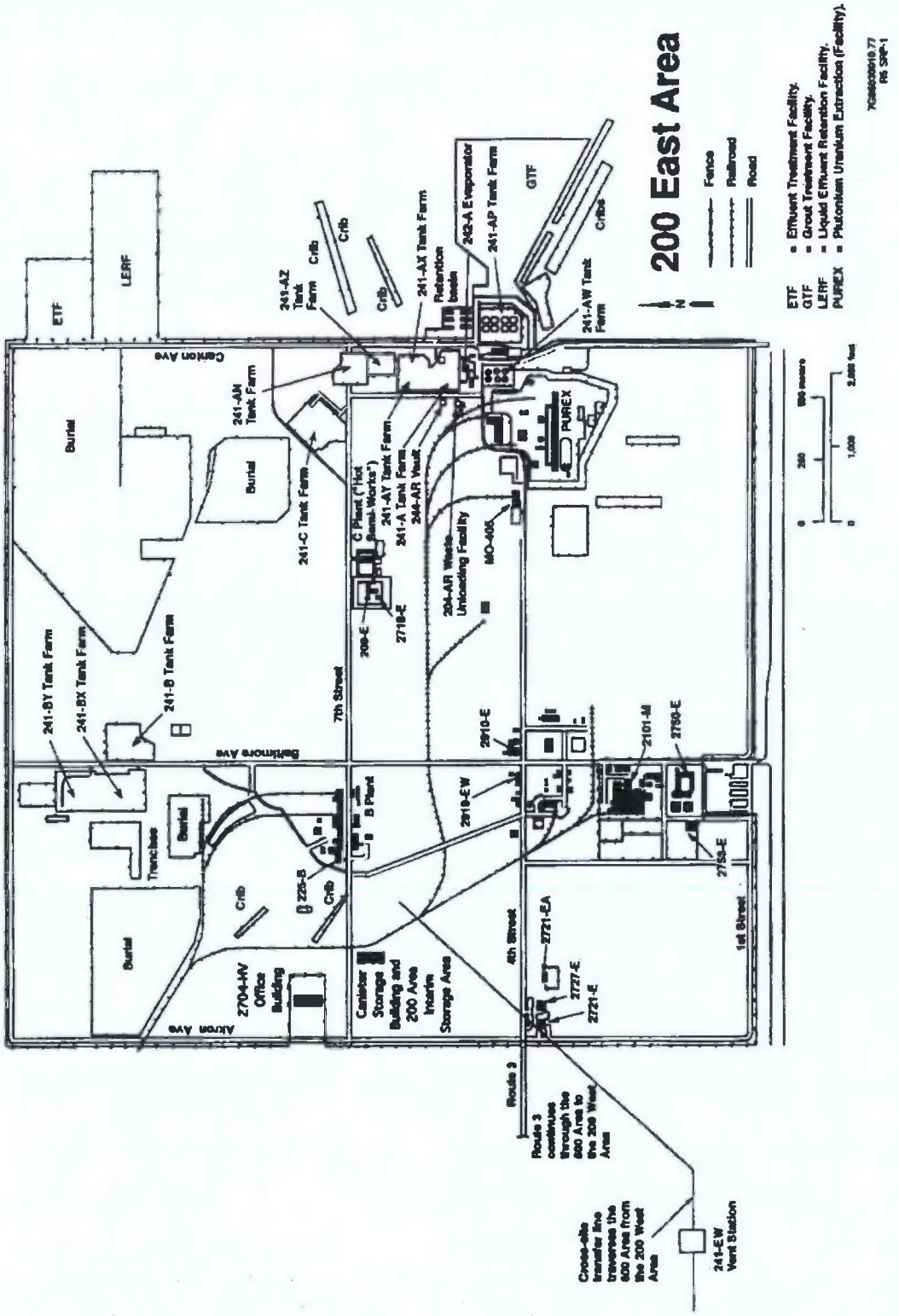


Figure A-3. 200 West Area.
- Official Use Only -

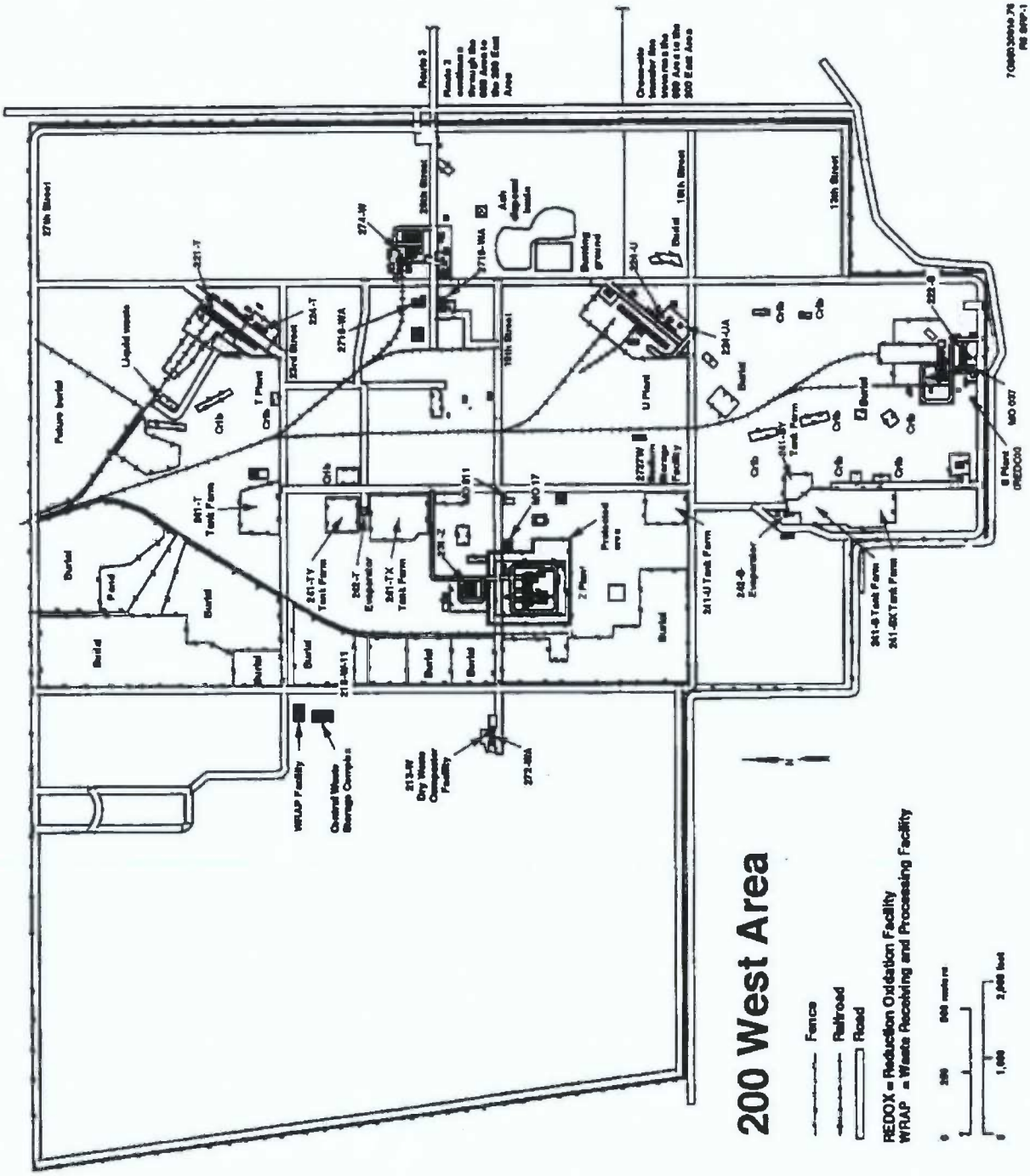


Figure A-4. 241-B Plant Site Plan.

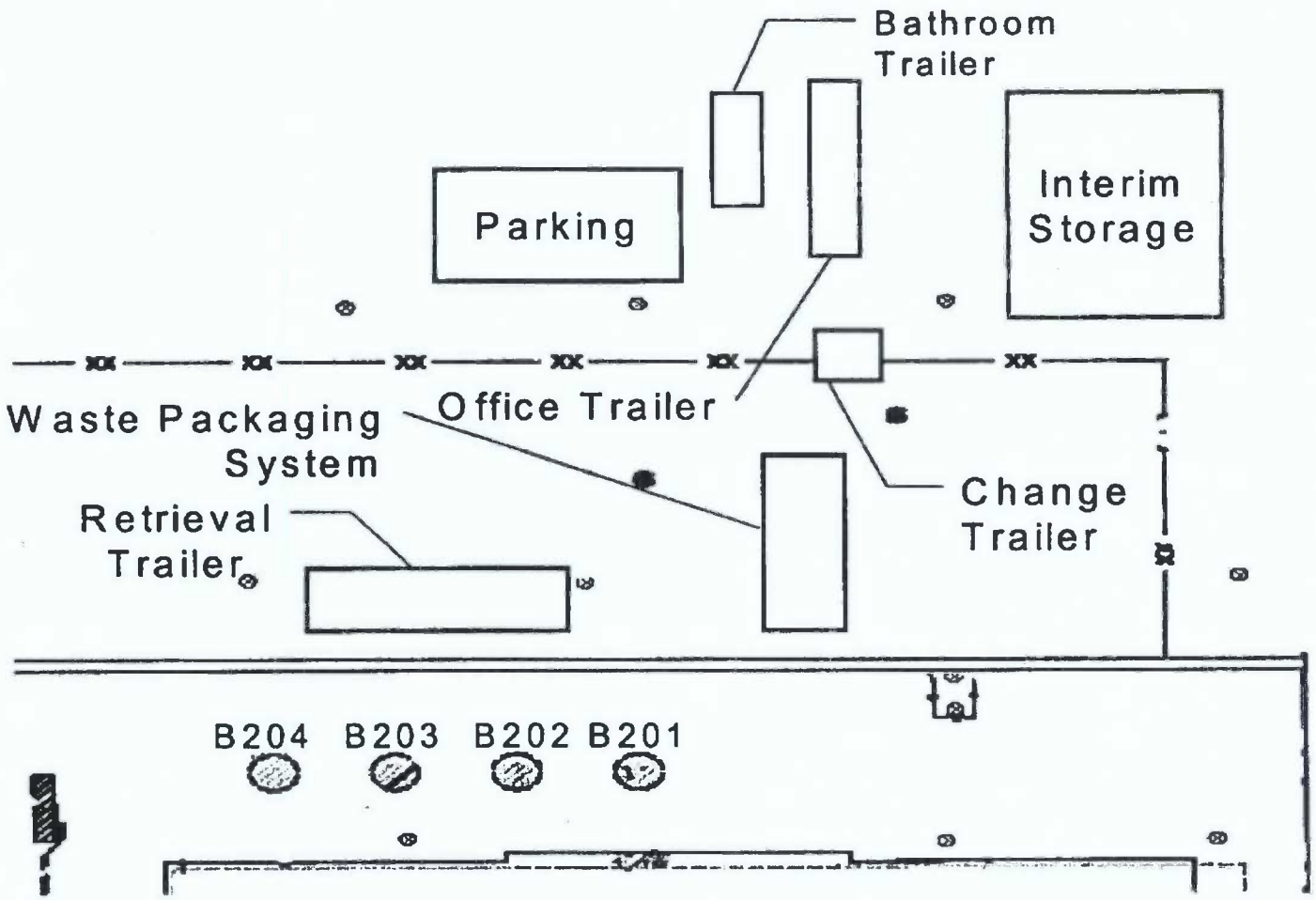


Figure A-5. 241-T Plant Site Plan.

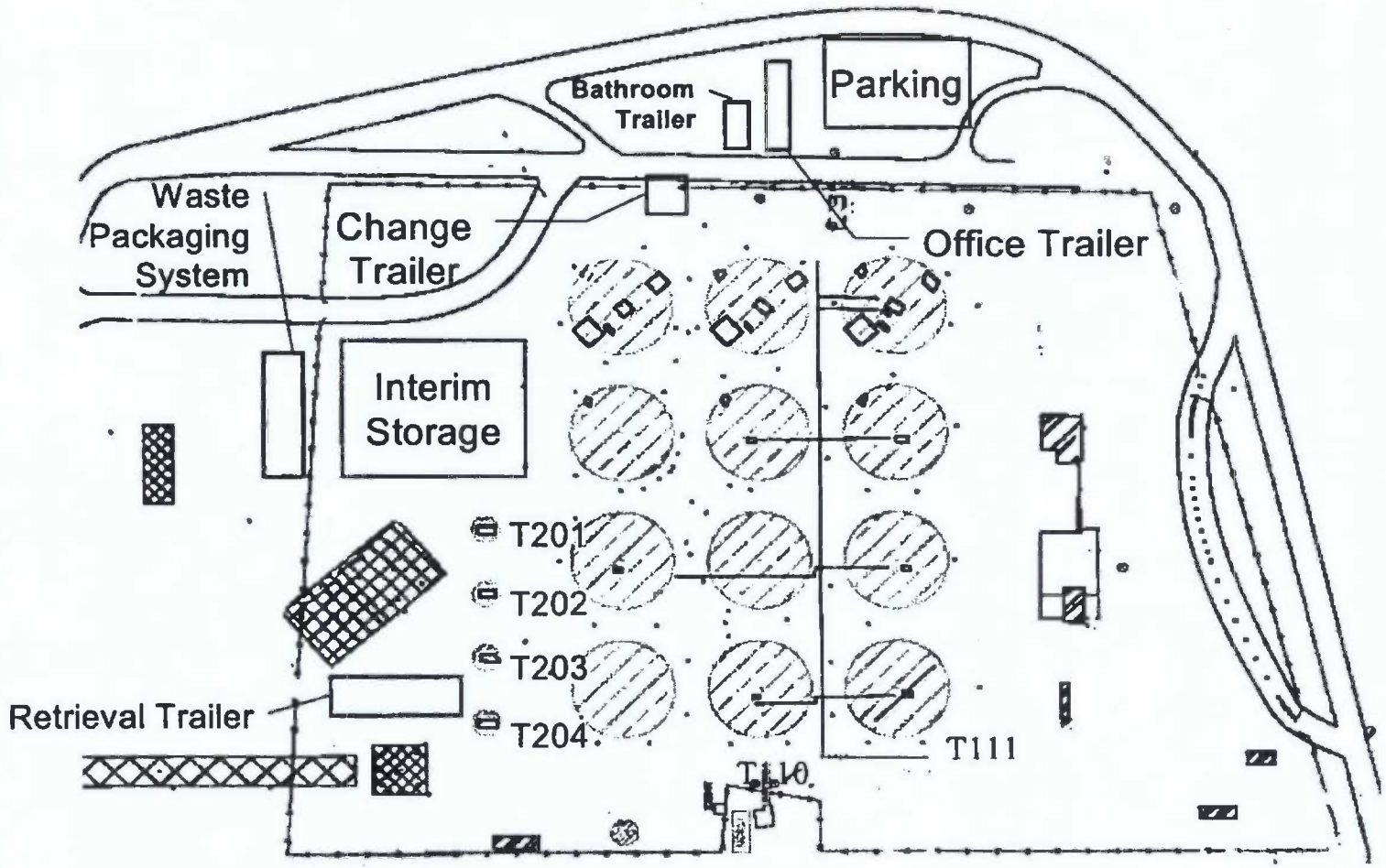
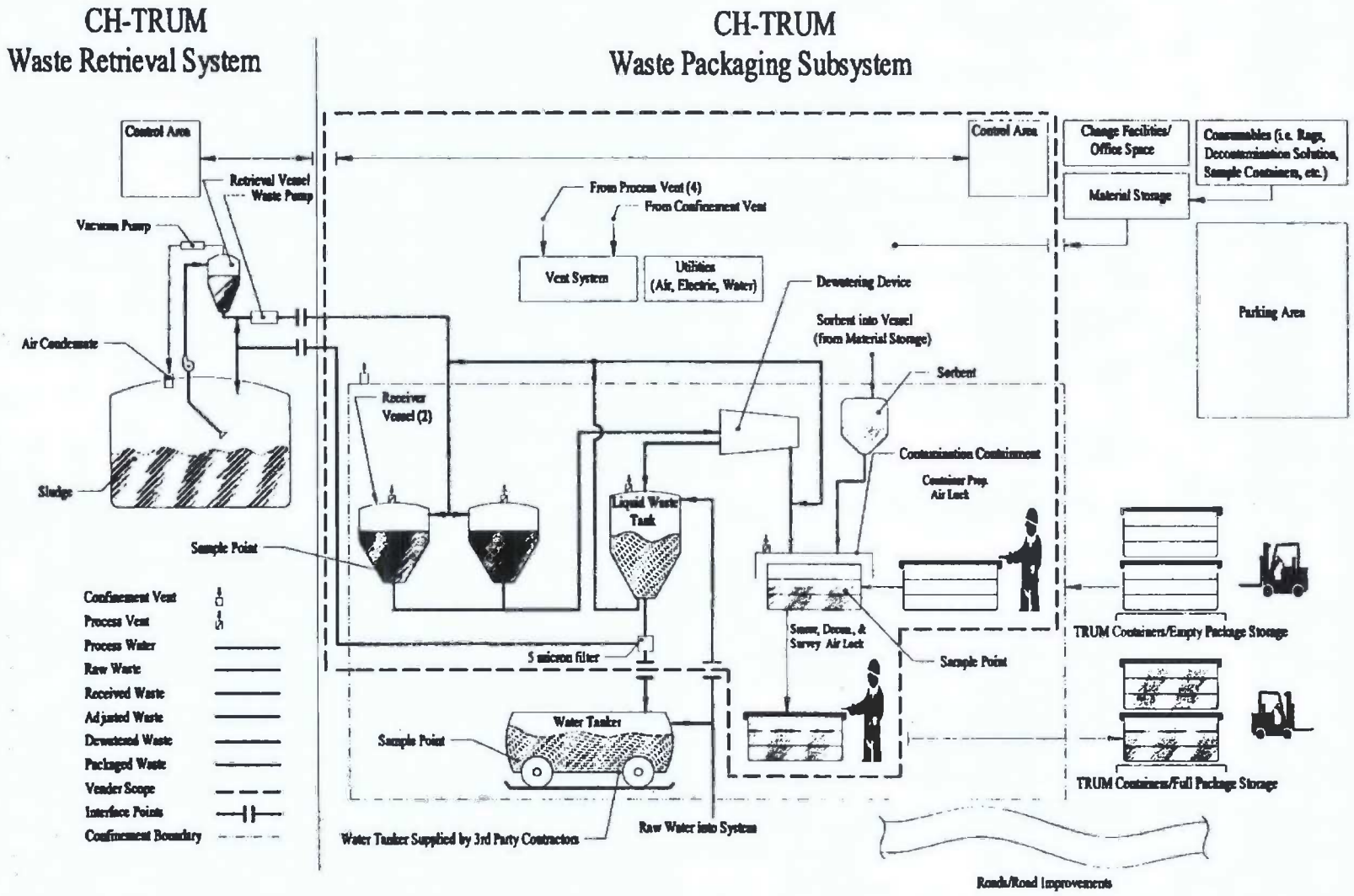


Figure A-6. CH-TRU Waste Packaging Subsystem Process Flow Diagram



APPENDIX B

FORMAL NOTICES OF VIOLATIONS AND/OR NOTICES OF PENALTIES

Hanford Site Compliance Violations and Response Summary

Tuesday, April 29, 2003

Received Date: April 14, 2003
Agency: U.S. Environmental Protection Agency
Summary:

The U.S. Environmental Protection Agency (EPA) issued a Notice of Penalty (NOP) letter to the U.S. Department of Energy (DOE), Richland Operations Office (RL) on April 3, 2003, that assesses a penalty against DOE RL for alleged violation of a Comprehensive Environmental Response Compensation and Liability Act (CERCLA) requirement that was agreed to within the Hanford Federal Facility Agreement and Consent Order (HFFACO). In this letter, EPA alleges that DOE RL failed to complete HFFACO Milestone M-34-08. This milestone required initiation of full scale sludge removal from the K East Basin located in the 100 K Area by December 31, 2002. The total penalty assessed for this alleged violation for the period through April 1, 2003, is $76,000.

Response(s):

DOE RL made a decision to not invoke dispute resolution.

Received Date: February 7, 2003
Agency: State of Washington Department of Ecology
Summary:

The State of Washington Department of Ecology (Ecology) issued a Notice of Non-Compliance letter to the DOE Office of River Protection (ORP) and CH2M HILL Hanford Group, Inc. (CHG) on February 6, 2003, that documents their concerns regarding the inspection and repair of leak detection equipment associated with AY, AZ, and SY Double-Shell Tank (DST) Farms. Ecology alleged that leak detection equipment associated with the AY, AZ, and SY Tank Farms has not been inspected or maintained in accordance with applicable Washington Administrative Code (WAC) or Code of Federal Regulations requirements. This Notice of Non-Compliance identified three alleged violations and one concern.

Response(s):

DOE and its contractors have agreed to corrective measures with Ecology. Corrective actions are in progress.

Received Date: December 10, 2002
Agency: State of Washington Department of Department of Ecology
Summary:

Ecology issued a Notice of Non-Compliance letter to RL, ORP, and CHG on December 3, 2002, that documents their concerns regarding the filling of tank farm dilution tanks. Ecology alleged that activities associated with the filling of tank farm dilution tanks and State Waste Discharge Permit (ST 4508) are not in compliance with the provisions of the State of Washington Water Pollution Control Law, Chapter 90.48 Revised Code of Washington (RCW), as amended; the Federal Water Pollution Control Act (The Clean Water Act), and per WAC 173-216-020 groundwater requirements. The Notice of Non-Compliance identified two alleged violations.

Response(s):

DOE and its contractors have chosen to not submit requested Certificates of Compliance.

DOE letter 03-ED-024, dated February 7, 2003, addresses the two corrective measures and provides a status for each.

Ecology letter, dated March 21, 2003, provides Ecology's acceptance of documentation submitted in response to the Notice of Non-Compliance. Ecology considers the inspection/investigation closed.

Received Date: August 13, 2002
Agency: State of Washington Department of Ecology
Summary:

Ecology issued a Notice of Non-Compliance letter to ORP on August 8, 2002, that documents their concerns regarding the leak detection system associated with temporary transfer lines used at the Single-Shell Tank (SST) Farms. Ecology alleged that the leak detection system associated with temporary transfer lines used at the SST Farms does not meet the WAC 173-303-400, Interim Status Facility Standards. The Notice of Non-Compliance identified two alleged violations and two concerns.

Response(s):

ORP letter 02-OMD-059, dated August 29, 2002, transmitted a request for extension of the corrective measures due dates to Ecology. An extension to October 23, 2002, was requested for Corrective Measure 1 and December 20, 2002, for Corrective Measure 2.

Ecology letter, dated September 18, 2002, grants the requested extension.

ORP letter 02-ED-039, dated December 17, 2002, transmitted the requested temporary transfer line management plan to Ecology.

Received Date: February 19, 2002
Agency: U.S. Environmental Protection Agency
Summary:

EPA issued a letter to the DOE on February 11, 2002, regarding waste management practices at the 233-S Plutonium Concentration Facility. The problem originated from the calibration and use of non-destructive assay (NDA) equipment for the designation of waste as low-level or transuranic waste. The letter claims that there was a "Failure to comply with waste designation requirements in the approved Sampling and Analysis Plan. The use of NDA data for designation was discussed during the Sampling and Analysis Plan development, but the plan was never revised to allow the use of NDA data for waste designation." There were no actions, fines, or penalties associated with this Notice of Violation (NOV).

Response(s):

None to date.

Received Date: October 17, 2001
Agency: Washington State Department of Health
Summary:

A NOV and Compliance Order was received from the Washington State Department of Health (WDOH) on October 15, 2001. WDOH alleges that the DOE prime contractor Fluor Hanford, Inc. (FHI) is in violation of WAC 246-247-040(4), which states that all existing emission units shall utilize As Low As Reasonably Achievable Control Technology.

DOE and its contractors were required in AIR 01-505 to develop procedures for indication devices for all emission units to ensure they "are monitored, trended and evaluated for changing conditions that may indicate abatement controls are not operating as designed."

It is alleged that DOE and several of its contractors did not comply by the required August 17, 2001 deadline. An additional 30 days were given to comply (AIR 01-811). The response provided to WDOH on October 1, 2001, contained a FHI management directive requiring compliance by the projects.

It is alleged that the project procedures were not provided until WDOH requested them on October 3, 2001. WDOH alleges that these procedures did not accomplish company wide compliance to the WDOH requirements or to FHI's own management directive.

Response(s):

RL letter 02-RCA-035, dated November 1, 2001, provided a response to the NOV and Compliance Order.

AIR 01-1105, dated November 15, 2001, provided WDOH closure of issues associated with letter AIR 00-1008.

Received Date: March 27, 2001
Agency: State of Washington Department of Ecology
Summary:

Ecology issued a NOP on March 26, 2001, in response to the identification of alleged waste management violations associated with the storage of a potentially shock sensitive chemical (trade name Collodion) as either waste or product in the 222-S Laboratory Complex, Waste Sampling and Characterization Facility, and Plutonium Finishing Plant (PFP) laboratories. The NOP levies a penalty of $57,800 against the DOE and FHI.

Response(s):

RL letter 01-RCA-245, dated April 9, 2001, transmitted an Application for Relief from Penalty No. 01NWPKW-2467 to Ecology.

Ecology letter, dated July 26, 2001, provides Ecology's rejection of the DOE Application for Relief from NOP No. 01NWPKW-2467. The Ecology letter identifies that the penalty described in NOP No. 01NWPKW-2467 is due and payable within 30 days of receipt of the letter.

RL letter 01-RCA-401, dated August 10, 2001, clarified the date of receipt of Ecology's letter. Due to problems encountered with the original letter, it was necessary for Ecology to reissue the letter. RL received the reissued letter on August 1, 2001. It is RL's position that the 30 day clock starts on August 1, 2001.

Received Date: March 26, 2001
Agency: Washington State Department of Health
Summary:

A NOV and Compliance Order was received from the WDOH on March 23, 2001. WDOH alleges that DOE failed to properly notify WDOH following a continuous air monitor alarm in the 291-Z-1 stack of the PFP on February 23, 2001. The NOV and Compliance Order requires DOE to propose to WDOH a corrective action to ensure this does not recur. The response is due within 60 days of the date of the WDOH letter. WDOH also posed a number of questions regarding the extent and nature of the release, as well as decisions that were made during and after the event.

Response(s):

RL letter 01-RCA-267, dated May 2, 2001, provided a response to the NOV and Compliance Order.

Received Date: June 27, 2000
Agency: U.S. Environmental Protection Agency
Summary:
EPA issued a letter to DOE and Bechtel Hanford, Inc. (BHI) on June 13, 2000, regarding waste management practices at multiple operable units. The letter claims that DOE and BHI violated CERCLA requirements agreed to in the HFFACO with respect to waste management practices at the 100-F, 100-K, 100-BC, 200-ZP-1, and 300-FF-2 operable units.

Response(s):

E mail, dated June 30, 2000, from Karen Hornbuckle, BHI, documented that the requirement to obtain EPA approval for all Investigation Derived Waste (IDW) shipped to Environmental Restoration Disposal Facility (ERDF) has been implemented.

RL letter 00-OSS-490, dated September 22, 2000, provided a point-by-point response to Ecology's letter of June 13, 2000.

RL letter 01-RCA-092, dated December 29, 2000, documented that the seven drums had been removed from the biosite and properly dispositioned. There was no record or evidence of any releases at the drum storage sites.

Received Date: June 22, 2000
Agency: State of Washington Department of Ecology
Summary:

Ecology issued Administrative Order No. 00NWPKW-1251 on June 13, 2000. The Administrative Order requires the DOE and CHG to comply with Chapter 70.105 RCW, the Hazardous Waste Management Act, Chapter 173-303 WAC, by reference Chapter 40, Code of Federal Regulations, and certain actions described as they apply to determining the integrity of the DST System.

Response(s):

RL letter 00-ORL-065, dated June 20, 2000, RL and ORP gave notice of their election to exercise their dispute resolution rights under Article VIII of the HFFACO.

Ecology letter, dated June 27, 2000, provided Ecology's determination that Administrative Order No. 00NWPKW-1251 is not subject to dispute resolution within the HFFACO.

ORP letter 00-OSD-080, dated July 19, 2000, submitted a Statement of Dispute signifying elevation of the issue to the Inter-Agency Management Integration Team.

ORP letter 00-OSD-108, dated September 18, 2000, submitted information in response to Actions 1.A, 1.B, 1.C, 2, 3, and 4.

Ecology letter, dated November 2, 2000, provided Ecology's identified deficiencies with DOE's September 18, 2000, submittal. The September 18, 2000, submittal addressed Action Items 1.A, 1.B, 1.C, 2, 3, and 4 of Administrative Order No. 00NWPKW-1251. Actions resulting from Ecology's letter will be tracked under the Environmental Actions Tracking System Item 20001102-ECL-LET-REQ. Ecology's letter also changed the due date for submittal of information required by Action 5 of the Administrative Order from December 16, 2000, to December 18, 2000.

ORP letter 00-OSD-175, dated December 18, 2000, submitted information required by Action Item 5 of the Administrative Orders.

ORP letter 00-OSD-177, dated December 28, 2000, requests Ecology's formal concurrence with tanks selected for ultrasonic testing in Fiscal Year (FY) 2001.

Ecology letter, dated January 24, 2001, provided Ecology's concurrence with tanks selected for ultrasonic testing in FY 2001.

Ecology letter, dated April 23, 2001, provided Ecology's acknowledgement of receipt of information required by Items 1A, 3, 4, and 5 of the Administrative Order. Ecology considers items 1A and 5 completed. Ecology also accepted the visual inspection plans submitted in accordance with the requirements of Items 3 and 4 of the Administrative Order. Ecology will consider these items completed provided the visual inspection plans are implemented within 60 days of the date of this letter

Received Date: June 20, 2000
Agency: State of Washington Department of Ecology
Summary:

Ecology issued NOP Incurred and Due No. 00NWPKW-1249 on June 13, 2000. The NOP assesses a penalty against DOE in the amount of $200,000 under the provisions of Article IX, Stipulated Dangerous Waste Penalties, within the HFFACO. This NOP was assessed for failure to meet the provisions of Article VII within the HFFACO, with respect to completion of major HFFACO Milestone M-32.

Response(s):

RL letter 00-ORL-065, dated June 20, 2000, RL and ORP gave notice of their election to exercise their dispute resolution rights under Article VIII of the HFFACO.

Ecology letter, dated June 27, 2000, provided Ecology's determination that NOP Incurred and Due No. 00NWPKW-1249 is subject to dispute resolution as provided by Article VIII of the HFFACO.

ORP letter 00-OSD-080, dated July 19, 2000, submitted a Statement of Dispute signifying elevation of the issue to the Inter-Agency Management Integration Team.

Ecology letter, dated August 24, 2000, provided Ecology's Final Determination in the matter pursuant to HFFACO Part Two, Article VIII, Paragraph 30 (D). This Final Determination was issued solely for resolution of disputes brought forth by DOE in relation to NOP Incurred and Due No. 00NWPKW-1249. This Final Determination demands immediate payment of the entire penalty amount described in NOP Incurred and Due No. 00NWPKW-1249 upon receipt of the Final Determination. This action will be tracked separately.

Received Date: June 16, 2000
Agency: State of Washington Department of Ecology
Summary:

Ecology issued Administrative Order No. 00NWPKW-1250 on June 13, 2000. The Administrative Order requires the DOE and CHG to comply with Chapter 70.105 RCW, the Hazardous Waste Management Act, Chapter 173-303 WAC, by reference Chapter 40, Code of Federal Regulations, and certain actions described as they apply to determining the integrity of the DST System.

Response(s):

RL letter 00-ORL-065, dated June 20, 2000, RL and ORP gave notice of their election to exercise their dispute resolution rights under Article VIII of the HFFACO.

Ecology letter, dated June 27, 2000, provided Ecology's determination that Administrative Order No. 00NWPKW-1250 is not subject to dispute resolution within the HFFACO.

ORP letter 00-OSD-080, dated July 19, 2000, submitted a Statement of Dispute signifying elevation of the issue to the Inter-Agency Management Integration Team.

ORP letter 00-OSD-108, dated September 18, 2000, submitted information in response to Actions 1.A, 1.B, 1.C, 2, 3, and 4.

Ecology letter, dated November 2, 2000, provided Ecology's identified deficiencies with DOE's September 18, 2000, submittal. The September 18th submittal addressed Action Items 1.A, 1.B, 1.C, 2, 3, and 4 of Administrative Order No. 00NWPKW-1250. Actions resulting from Ecology's letter will be tracked under EATS Item 20001102-ECL-LET-REQ. Ecology's letter also changed the due date for submittal of information required by Action 5 of the Administrative Order from December 16, 2000, to December 18, 2000.

ORP letter 00-OSD-175, dated December 18, 2000, submitted information required by Action Item 5 of the Administrative Orders.

ORP letter 00-OSD-177, dated December 28, 2000, requests Ecology's formal concurrence with tanks selected for ultrasonic testing in FY 2001.

Ecology letter, dated January 24, 2001, provided Ecology's concurrence with tanks selected for ultrasonic testing in FY 2001.

Ecology letter, dated April 23, 2001, provided Ecology's acknowledgement of receipt of information required by items 1A, 3, 4, and 5 of the Administrative Order. Ecology considers items 1A and 5 completed. Ecology also accepted the visual inspection plans submitted in accordance with the requirements of Items 3 and 4 of the Administrative Order. Ecology will consider these items completed provided the visual inspection plans are implemented within 60 days of the date of this letter.

Received Date: March 20, 2000
Agency: Washington State Department of Health
Summary:

WDOH has issued a NOV and Compliance Order as authorized by WAC 246-247-100(a) and RCW 70.94.332 for actions taken at the 244-AR Vault. The 244-AR Vault is located in the 200 East Area and serves as a waste transfer station. WDOH identified three violations and three compliance orders.

Response(s):

ORP letter 00-ESHQ-007, dated May 3, 2000, submitted response documentation demonstrating the three compliance orders have been completed. Compliance orders 1 and 2 were completed through revisions to HNF-IP-0842. The training requirement was accomplished by Tank Farm personnel attending training provided by WDOH. Compliance order 3 was met by submittal of a Notice of Construction (NOC) for Categorical Facility Entry and Surveillance on May 3, 2000.

Received Date: March 7, 2000
Agency: U.S. Environmental Protection Agency
Summary:

EPA has assessed penalties against RL in response to violations of CERCLA requirements that were agreed to within the HFFACO. The penalties assessed regard waste management practices at the 221-U Facility located in the 200 West Area. The two identified violations were described previously in a letter from the EPA to RL issued on November 17, 1999. The violations concern the failure to have an approved Waste Control Plan (WCP) and failure to sample waste per the approved Sampling and Analysis Plan (SAP). The total penalty assessed for both violations equals $55,000.

Response(s):

RL letter 00-OSS-330, dated May 5, 2000, provided written confirmation to the EPA that the $55, 000 penalty was paid in full on April 18, 2000. Remittance was made electronically to the EPA Superfund Accounting Office in Pittsburgh, Pennsylvania.

Received Date: January 14, 2000
Agency: State of Washington Department of Ecology
Summary:

Ecology has formally denied the Application for Relief from Penalty submitted by RL and BHI on December 1, 1999, in response to Penalty #99NWKW-21 that was issued on November 17, 1999. The penalty was issued for failure to adequately designate waste stored in the 271-U 90-day accumulation area. RL and BHI may either pay the penalty in full or appeal the denial to the Pollution Control Hearings Board (PCHB).

Response(s):

RL letter 00-ERD-053, dated February 9, 2000, documented remittance of $9,700 to the Fiscal Cashier, Ecology fulfilling the denial of the Application for Relief from Penalty received on January 14, 1999. Payment of the penalty does not constitute an admission by either or both respondents of the allegations of the NOP, liability under Washington State law, or of jurisdiction by Ecology over the specific subject matter of the NOP.

Received Date: November 18, 1999
Agency: U.S. Environmental Protection Agency
Summary:

EPA informed the DOE and BHI of CERCLA violations identified in letter issued November 17, 1999. Two violations were identified following an inspection of the 271-U 90-Day Accumulation area conducted by the Ecology on September 16, 1999. The 271-U 90-day accumulation area is located at U Plant in the 200 West Area.

Response(s):

RL letter 00-OSS-074, dated November 24, 1999, submitted documentation seeking to confirm a request for a 20-day extension to the due date for the requested written action plan. The request was originally made verbally during a meeting with a representative of the EPA on November 19, 1999. The request sought an extension of 20 days to December 17, 1999.

RL letter 00-OSS-097, dated December 16, 1999, submitted an Action Plan Identifying Corrective Actions and Responses to the EPA and Ecology in accordance with the approved extension. The Action Plan contained responses, corrective actions taken, and follow-up corrective actions for each violation. Responses were provided for each concern.

Ecology letter dated February 23, 2000, acknowledged receipt and accepts the Action Plan submitted by RL on December 16, 1999. Ecology has accepted the Action Plan as qualified by identified conditions, and considers the violation addressed and inspection closed.

E mail, dated April 17, 2000, documented completion of all six compliance commitments made in accordance with the corrective action plan submitted on December 16, 1999 (RL letter #00-

OSS-097): the 221-U Facility Canyon Disposal Initiative (CDI) SAP was revised on February 2, 2000, to include management of unknowns. EPA and Ecology approved the revision - three active WCPs were revised on February 2, 2000, to include treatment of waste as a standard practice as agreed to in regulatory meetings, approximately 14 inactive WCPs will be reviewed and revised prior to generation of additional waste - designation procedure BHI-FS-03, WOO2, was revised on February 7, 2000, to identify criteria used to evaluate process knowledge and now includes identification of all process knowledge documentation - training to identify and explain procedure changes was conducted on January 27, 2000, - modifications to the 221-U Facility CDI WCP were provided to the EPA and Ecology for review and were approved on February 2, 2000, - procedures were modified on February 7, 2000, to prohibit the use of standardized or boilerplate language in the development of Waste Profiles.

Received Date: November 18, 1999
Agency: State of Washington Department of Ecology
Summary:

Ecology assessed a penalty of $9,700 against the DOE and BHI on November 17, 1999, under the provisions of the RCW 70.105.080. The penalty is the result of findings associated with an Ecology inspection of the 271-U 90-day accumulation area conducted on September 16, 1999.

Response(s):

RL letter 00-OSS-085, dated December 1, 1999, submitted an Application for Relief from Penalty in accordance with the provisions outlined original NOP issued by Ecology on November 17, 1999.

Ecology letter, dated January 12, 2000, transmitted a formal denial from relief from Penalty #99NWPKW-21/#99NWPKW-22 for reasons stipulated. RL and BHI have 30 days to either appeal the denial to the PCHB or pay the penalty in full. These actions will be tracked as a separate item.

RL letter 00-ERD-053, dated February 9, 2000, documented remittance of $9,700 to the Fiscal Cashier, Ecology fulfilling the denial of the Application for Relief from Penalty received on January 14, 1999. Payment of the penalty does not constitute an admission by either or both respondents of the allegations of the NOP, liability under Washington State law, or of jurisdiction by Ecology over the specific subject matter of the NOP.

Ecology letter dated February 23, 2000, acknowledged receipt and accepts the Action Plan submitted by RL on December 16, 1999. Ecology has accepted the Action Plan as qualified by identified conditions, and considers the violation addressed and inspection closed.

Received Date: July 20, 1999
Agency: South Carolina Department of Health
Summary:

Following an investigation by the South Carolina Department of Health and Environmental Control, violation of state and federal regulations were identified. On May 20, 1999, a CNS-1-13G, Type B shipping cask was discovered upon receipt at the Chem-Nuclear Systems (CNS) at Barnwell, SC, to have removable contamination levels exceeding U.S. Department of Transportation limits. In addition, another incident involving a similar cask shipment on November 24, 1999, an unmanifested sample container rack and liquids were discovered in the cask upon receipt. Both casks had been shipped to CNS for maintenance work.

Response(s):

99-SFD-152, dated July 26, 1999, submitted RL corrective documentation to the South Carolina Department of Health as requested. The corrective action documentation included measures to prevent recurrence.

Letter, dated August 23, 1999 - The South Carolina Department of Health and Environmental Control reviewed the corrective measures submitted and found them to be adequate to preclude recurrence of the discrepancy that resulted in the NOV. This item is considered closed.

Received Date: February 16, 1999
Agency: U.S. Environmental Protection Agency
Summary:

EPA and Ecology conducted a Multi-Media Inspection (MMI) of the Hanford Site from May through July 1998. The inspection identified concerns that resulted in the issuance of three violations of Resource Conservation and Recovery Act (RCRA) regulations that include; storage without a permit, failure to make a hazardous waste determination, and failure to immediately amend a contingency plan. Civil penalties were assessed for each violation in the sum of $367,078.00.

Response(s):

RL letter 99-EAP-031, dated November 9, 1998, submitted supplemental information to the EPA in support of the multi-media inspection. The information consisted of supporting data regarding waste designation.

Meeting on February 19, 1999, - RL and contractor representatives meet informally with the EPA. The discussions were preliminary in nature and did not result in a settlement.

RL letter 99-OCC-0118, dated March 17, 1999, provided a formal response to the Complaint and included a request for hearing. In addition, the response identified defenses for each of the three counts and made a request for dismissal.

An administrative law judge has been appointed and a pre-hearing order has been issued. The pre-hearing discovery phase is to be completed by June 22, 1999. This requires that all documents used in the defense must be identified and available and a witness list will have to be prepared, which includes a brief summary on the witness testimony. In addition, a draft settlement is being prepared that will include potential supplemental environmental projects (SEP). Meetings are being held that started on Tuesday, April 27th, to start the pre-hearing effort.

EPA letter OEC-164, dated March 9, 2000, - transmitted the finalized EPA MMI inspection report to RL. The report contains reviews of RCRA, National Pollutant Discharge Elimination System, Toxic Substances Control Act of 1976, polychlorinated biphenyl, Clean Air Act (CAA), Emergency Planning and Community Right to Know Act, National Emission Standards for Hazardous Air Pollutants/Asbestos, and Underground Storage Tanks. The EPA is not planning any formal enforcement activities for the above portions of the inspection with the exception of RCRA.

A prehearing was held in September 1999. During the prehearing, the EPA withdrew Count III, failure to have a contingency plan, with prejudice, from the complaint. A court date has been set for June 2000 and the hearing will be held in Richland, Washington.

EPA letter ORC-158, dated October 12, 2000 - documents the terms of the settlement of the Multimedia Inspection matter in a Consent Agreement and Final Order (CAFO). The CAFO requires payment of a civil penalty, performance of two SEPs, and the performance of specified compliance activities.

Ecology letter, dated June 13, 2002, provides Ecology's acceptance of the closure of the 200 East Area Pipe Laydown Yard Accumulation Area, and considers the requirements of the CAFO (RCRA-10-99-0106) Paragraph 23, Compliance Activities fulfilled.

Received Date: January 7, 1999
Agency: State of Washington Department of Ecology
Summary:

Ecology has formally denied the Application for Relief from Penalty (98NM-007) submitted on October 3, 1997. Ecology issued Penalty #97NM-248 on September 16, 1997, in the amount of $110,000. The penalty was assessed as the result of the failure to properly manage chemicals and for the inadequate response to the subsequent chemical release from the Plutonium Reclamation Facility.

Response(s):

An appeal was filed with the PCHB with a pre-hearing originally scheduled for March 24, 1999. Ecology has requested the pre-hearing meeting be moved to April 2, 1999. The appeal completes the action associated with this item.

Settlement Agreement, dated July 7, 1999, - The Attorney General of Washington offered RL, Fluor Daniel Hanford, Inc. (FDH), and Babcock and Wilcox Hanford Company to enter into a Settlement Agreement resolving the NOP 97NM-248 issued by Ecology September 16, 1997. The proposed Settlement Agreement stipulates duration of agreement, innovative settlement payment, enforcement during the term of agreement, and dispute resolution. PCHB approval was obtained on July 7, 1999, making the Settlement Agreement effective.

Received Date: October 23, 1998
Agency: Attorney General
Summary:

The Attorney General of Ecology have agreed to stay of Administrative Order No. 98NW-009 issued on September 24, 1998, and addresses compliance with Federal and State hazardous waste regulations for the management of mixed waste in tanks. The stay is effective until January 29, 1999, and has been issued to aid in the process of settlement of the issues in the appeal of that order to be filed by the Appellants to the PCHB.

Response(s):

Settlement Agreement No. PCHB 98-249 and 98-250, dated February 26, 1999, - A settlement agreement was reached between Ecology, RL, FDH, and Lockheed Martin Hanford Corporation (LMHC) on February 26, 1999. The agreement suspended Administrative Order 98NW-009, pending signature of the agreement by the PCHB. Signature by the PCHB and Administrative Appeals Judge was secured on March 15, 1999.

Ecology letter, dated October 6, 2000, closes out Ecology's March 24, 1998, inspection of the SY Tank Farm.

Ecology letter, dated April 6, 2001, provides Ecology's acceptance of the DST Emergency Pumping Guide that was submitted on March 6, 2001. Ecology considers all actions associated with the July 8, 1998, Notice of Correction closed.

Received Date: September 24, 1998
Agency: State of Washington Department of Ecology
Summary:

Ecology formally denied the Application for Relief from Penalty 98NW-007 issued to DOE, FDH, and LMHC on July 23, 1998. Penalty 98NW-007 was levied following the issuance of a NOC in response to a compliance inspection at the SY Tank Farm on March 24, 1998. The Application for Relief was received by Ecology on August 7, 1998. A review of the application revealed no new or extraordinary information relevant to dismissing the penalty.

Response(s):

October 23, 1998; The denial for Application for Relief from Penalty 98NW-007 was appealed by RL to the PCHB. November 4, 1998, meeting; RL, FDH, and LMHC proposed a reasonable settlement of the Penalty to Ecology. Ecology responded to indicate a counter proposal would be forthcoming.

Settlement Agreement, dated February 26, 1999, - A settlement agreement was reached between Ecology, RL, FDH, and LMHC on February 26, 1999. The agreement suspended Administrative Order 98NW-009, pending signature of the agreement by the PCHB. Signature by the PCHB and Administrative Appeals Judge was secured on March 15, 1999.

Ecology letter, dated October 6, 2000, closes out Ecology's March 24, 1998, inspection of the SY Tank Farm.

Ecology letter, dated April 6, 2001, provides Ecology's acceptance of the DST Emergency Pumping Guide that was submitted on March 6, 2001. Ecology considers all actions associated with the July 8, 1998, NOC closed.

Received Date: September 24, 1998
Agency: State of Washington Department of Ecology
Summary:

Ecology issued Administrative Order No. 98NW-009 on September 24, 1998. The Administrative Order requires DOE, FDH, and LMHC to comply with Chapter 70.105 RCW, Chapter 173-303 WAC, by reference Chapter 40, Code of Federal Regulations, and certain actions described as they apply to the management of waste at SY Tank Farm.

Response(s):

State of Washington, Attorney General issued a stay to Administrative Order 98NW-009 on October 23, 1998, following an appeal of that order by RL to the PCHB the same day. The stay remains in effect until January 29, 1999. The stay was issued to aid the process of settlement of the issues in the appeal of that order to be filed by the Appellants to the PCHB.

This item has been closed per Settlement Agreement No. PCHB 98-249 and 98-250. The Settlement Agreement was reached between Ecology, RL, FDH, and LMHC on February 26, 1999. The agreement suspended Administrative Order 98NW-009, pending signature of the agreement by the PCHB. Signature by the PCHB and Administrative Appeals Judge was secured on March 15, 1999.

Ecology letter, dated October 6, 2000, closes out Ecology's March 24, 1998, inspection of the SY Tank Farm.

Ecology letter, dated April 6, 2001, provides Ecology's acceptance of the DST Emergency Pumping Guide that was submitted on March 6, 2001. Ecology considers all actions associated

with the July 8, 1998, NOC closed.

Received Date: July 23, 1998
Agency: State of Washington Department of Ecology
Summary:

Ecology assessed a penalty (98NW-007) against the RL, FDH, and LMHC in the amount of $75,600 under the provisions of the RCW 70.105.080. RL, FDH, and LMHC failed to provide a leak detection system for DSTs SY-101, 102, and 103 capable of detecting a leak from the primary or secondary structure of these tanks within 24 hours.

Response(s):

98-EAP-425, dated August 6, 1998, RL submitted an Application for Relief from Penalty 98NW-007 in accordance with protocols identified in the original NOP.

September 24, 1998; Ecology received and reviewed the Application for Relief from Penalty and responded with a Notice of Denial of Application for Relief from Penalty. The denial was based on the view held by Ecology and no new or extraordinary information relevant to dismissing the penalty was provided in the application. The option to appeal the denial to the PCHB was provided to the petitioners.

October 23, 1998; The denial for Application for Relief from Penalty 98NW-007 was appealed by RL to the PCHB.

Settlement Agreement No. PCHB 98-249 and 98-250, dated February 26, 1999, - A settlement agreement was reached between Ecology, RL, FDH, and LMHC on February 26, 1999. The agreement in part, stipulated the resolution of penalty 98NW-007, pending signature of the agreement by the PCHB. Signature by the PCHB and Administrative Appalls Judge was secured on March 15, 1999.

Ecology letter, dated October 6, 2000, closes out Ecology's March 24, 1998, inspection of the SY Tank Farm.

Ecology letter, dated April 6, 2001, provides Ecology's acceptance of the DST Emergency Pumping Guide that was submitted on March 6, 2001. Ecology considers all actions associated with the July 8, 1998, NOC closed.

Received Date: July 10, 1998
Agency: Washington State Department of Health
Summary:

WDOH has issued a NOV and NOC for violations of radioactive air emissions regulations at the 296-A-42 emission unit. The violation involves the intentional bypass of required controls and

the lack of any notification made to WDOH. Two violations and three corrective measures issued as compliance orders have been identified.

Response(s):

98-EAP-465, dated August 21, 1998, submitted required documentation to address Compliance Order #1.

98-EAP-422, dated August 10, 1998, submitted required documentation to address Compliance Orders #2 and #3.

AIR 99-105, dated January 13, 1999, provided written closure of the NOV following a review of the information submitted by RL.

Received Date: June 8, 1998
Agency: State of Washington Department of Ecology
Summary:

Ecology provided a 60-day NOI to sue RL for failing to meet SST Interim Stabilization milestone due dates. RL has the option to settle with a consent decree or proceed to trial.

Response(s):

February 22, 1999, following negotiations between senior DOE and State of Washington officials, the threatened law suit regarding the M-41-22 and M-41-23 HFFACOs milestones was cancelled and the parties have entered into a Consent Decree. The Consent Decree was issued by the Attorney General's Office on February 22, 1999, and contains provisions for a renegotiated schedule regarding the interim stabilization of SSTs.

Consent Decree closes this action.

Interim stabilization of the remaining SSTs will be renegotiated through the HFFACO.

Received Date: May 13, 1998
Agency: Washington State Department of Health
Summary:
WDOH found RL in violation of radioactive air emissions regulations in the operation of the Plasma Arc Furnace in the 324 Building. WDOH has issued a NOV and a NOC. RL conducted a project to treat neutron generators in the furnace during the week of April 13, 1998. RL shut down the stack tritium sampler for the duration of the project. RL did not submit a request for approval of periodic confirmatory monitoring to verify low emissions.

Response(s):

WDOH letter AIR 98-706, dated July 10, 1998, provided an extension to corrective action #1 based on a meeting held June 29, 19998, during which an extension agreement was reached. Evaluation of all NOCs is now due August 25, 1998.

98-EAP-441, dated August 20, 1998, RL submitted the required report in accordance with Compliance Order, #3. The report will be reviewed by WDOH to determine if revisions are needed for other Hanford Site NOCs. Due dates for any such revisions will be negotiated between RL and WDOH at that time.

Routine Technical Assistance Meeting, held on November 17, 1998, provided data to WDOH regarding 324 and 327 Building tritium measurements. WDOH was notified of the intent to discontinue tritium sampling associated with closure of the plasma arc furnace. Approval from WDOH was sought for this discontinuation. Closure of the furnace also eliminated the need for a NOC modification prior to further operation.

Communications Summary, dated January 19, 1999, summarized communications with WDOH personnel regarding a schedule for revising NOC discrepancies in accordance with the compliance order contained in the original NOV/NOC for the 324 Building.

99-EAP-260, dated April 20, 1999, submitted NOC updates to the EPA in response to the WDOH issued NOV/NOC. The updates were the result of a schedule developed by RL and WDOH to identify any discrepancies and resubmit applicable NOCs. The updated NOCs approved by WDOH were submitted for approval by the EPA.

99-EAP-261, dated April 27, 1999, submitted NOC updates to the EPA in response to the WDOH issued NOV/NOC. The updates were prepared in accordance with the Compliance Order issued as a part of the NOV/NOC and a schedule negotiated with WDOH. The updated NOCs were submitted for records purposes only.

AIR 99-609, dated June 9, 1999, provided formal closure of the NOV and Audit #555125 from WDOH.

Received Date: February 25, 1998
Agency: U.S. Environmental Protection Agency
Summary:

On February 25, 1998, the EPA issued a NOV to RL for violating the requirements defined in the ERDF Record of Decision. Three violations of the CAA and RCRA regulatory drivers were identified.

Response(s):

RL Letter 056862, dated March 20, 1998, submitted the required response to the corrective actions identified.

No response has been received from the EPA to date.

Received Date: November 13, 1997
Agency: State of Washington Department of Ecology
Summary:

Ecology issued Revised NOP Incurred and Due No. 97NM-139 on November 13, 1997. The NOP assesses a penalty against DOE in the amount of $90,000 under the provisions of the RCW 70.105.080. This Revised NOP was assessed for the alleged failure to maintain control of waste accumulated in the 222-S Laboratory Complex per WAC 173-303-200(2), satellite accumulation.

Response(s):

RL letter, dated December 12, 1997, transmitted a Notice of Appeal to the PCHB for the revised NOP.

The Stipulation and Agreed Order of Dismissal, PCHB 97-189, dated June 1, 2001, was issued to resolve the dispute over the Revised NOP Incurred and Due No. 97NM-139.

Received Date: September 16, 1997
Agency: State of Washington Department of Ecology
Summary:

In 1997, a chemical mixture stored for over a year in a tank located in the Plutonium Reclamation Facility underwent a spontaneous reaction rapidly generating sufficient pressure to violently rupture the tank. After concluding its investigation of the incident, Ecology served RL a NOP and NOC demanding payment of the sum of $110,000 for, inter alia, alleged violation of regulations prohibiting improper storage of hazardous waste. Corrective measures (CM) described at the end of the NOC letter were developed after the meetings regarding on-going actions being performed by DOE and its contractors.

Response(s):

In January 1998, Ecology performed a compliance inspection at the Plutonium Finishing Plant. It is DOE's understanding that Ecology intends to incorporate further discussion regarding the disposition of the items subject to CM 6 into closure actions to be taken following issuance of the Ecology compliance inspection report. While DOE has been waiting issuance of Ecology's compliance inspection report, DOE pursued field activities to disposition the remaining items. No report has been received concerning this Ecology inspection

On February 2, 1998, DOE transmitted a letter to Ecology identifying the remaining CMs and requested an extension date of July 1, 1998. On March 16, 1998, DOE sent a letter to Ecology supplying a status related to the disposition of the items identified in CM 4. This letter also transmitted the emergency preparedness documentation being submitted for the closure of CMs 1

and 2 for Ecology's review and comment.

On April 15, 1998, DOE submitted final documentation to close out CMs 1 and 2 that will become effective on July 1, 1998.

DOE responded to the NOP by filing with Ecology an Application for Relief from Penalty, which Ecology denied on January 7, 1999. DOE has 30 days from January 7, 1999, to appeal to the PCHB.

Ecology letter, dated August 8, 2000, - Ecology concurs that the six corrective measures have been met and DOE and its contractors completed measures required to settle penalty #97NM-248. Ecology considers these matters closed.

Received Date: April 30, 1997
Agency: State of Washington Department of Ecology
Summary:

Ecology issued NOP Incurred and Due No. 97NM-139 on April 28, 1997. The NOP assesses a penalty against DOE in the amount of $90,000 under the provisions of the RCW 70.105.080. This NOP was assessed for the alleged failure to maintain control of waste accumulated in the 222-S Laboratory Complex per WAC 173-303-200(2), satellite accumulation.

Response(s):

RL letter, dated May 13, 1997, provided the DOE, FDH, and Rust Federal Services of Hanford Inc. (RFSH) Application for Relief from Penalty 97NM-139.

Ecology letter, dated May 15, 1997, acknowledged Ecology's receipt of the Application for Relief from Penalty 97NM-139. In this letter, Ecology offered to meet with DOE, FDH, and RFSH prior to issuing a formal response to the Application for Relief.

Ecology letter, dated July 2, 1997, offered prospective dates to meet with DOE, FDH, and RFSH prior to issuing a formal response to the Application for Relief.

Ecology letter, dated July 15, 1997, identified that a written addendum to the Application for Relief will be required for Ecology to consider additional information relating to the DOE, FDH, and RFSH Application for Relief from Penalty 97NM-139.

Ecology letter, dated November 13, 1997, issued Revised NOP Incurred and Due No. 97NM-139 in the amount of $90,000 for alleged violations of WAC 173-303-200(2).

Received Date: July 24, 1996
Agency: State of Washington Department of Ecology
Summary:

Ecology performed an inspection of the 306-E Facility to follow up an Ecology inspection that occurred on September 14, 1995. One of the issues that Ecology had at that time concerned material being stored in two cabinets that contained what Ecology said appeared to be incompatible chemicals that could pose a threat to human health and the environment. Ecology issued a VCL on July 24, 1996, for storage of incompatible waste.

Ecology issued a formal NOP to DOE and Westinghouse Hanford Company (WHC) that included a $20,000 fine concerning storage of incompatible waste.

Response(s):

A formal response letter and payment of penalty was sent from WHC to Ecology on October 21, 1996. This enforcement action is considered closed.

On August 1, 1997, Ecology transmitted a letter of closure for the 306-E Facility stating that the corrective measures have been satisfied.

Received Date: March 6, 1996
Agency: State of Washington Department of Ecology
Summary:

Ecology issued a NOV (DE 96NM-033) to DOE alleging violation of WAC 173-400-141, -110, and -115 dealing with Prevention of significant deterioration (PSD) permitting, new source review, and new source performance standards under Washington's Clean Air Act. The NOV was issued on March 6, 1996. Ecology alleges that DOE is in violation of WAC 173-400-141 for failure to apply for and obtain the required state PSD permit and then operate the 300 Area boiler package without the permit, and in violation of WAC 173-400-115 for failure to meet new source performance standards for SO2 emission limits from the boiler. Construction of the 300 Area package boiler commenced in September 1989. Ecology determined that construction of the boiler constituted a major modification of the source subject to the PSD permit requirements. Additionally, the boiler has burned No. 6 fuel oil, and Ecology estimates that the SO2 emission rates exceed the NSPS's SO2 emission limits.

Response(s):

On August 12, 1996, Ecology transmitted their Agreed to Order to close this NOV. The Order proposes to close the NOV without fines or penalties if followed by DOE.

Received Date: February 8, 1996
Agency: Washington State Department of Health
Summary:

WDOH issued a NOV to RL after two field inspectors were denied access to portions of B Plant Complex emission units. WAC 246-247-100 requires facilities to "ensure all emission units are fully accessible to Department inspectors." The NOV required RL to resolve the denial of access problems.

Response(s):

On March 1, 1996, RL transmitted an interim response to WDOH. The response documented a meeting conducted between RL and WDOH representatives on February 27, 1996, during which an extended due date of 30 days was agreed to. On April 8, 1996, RL transmitted a final response to WDOH that addressed the requirements of the compliance order.

On April 23, 1996, WDOH responded to the April 8, 1996, submittal from RL. WDOH provided comments to the documentation submitted by RL and requested the comments be addressed before the issue could be closed. RL responded to the comments provided by WDOH and a verbal agreement was reached closing the violation.

Received Date: January 19, 1996
Agency: State of Washington Department of Ecology
Summary:
Ecology issued an NOP Incurred and Due (No. DE 96-NW-001) to DOE and BHI. The penalty was assessed based on a violation revealed from an investigation into dangerous waste management activities at the 183-H basins closure project. A $5,000 fine was assessed against DOE and BHI.

Response(s):

The penalty was paid and the NOP is considered closed.

Received Date: May 30, 1995
Agency: State of Washington Department of Ecology
Summary:

On May 30, 1995, Ecology issued a NOP Incurred and Due (No. DE 95NW-127) to DOE and Pacific Northwest National Laboratory after a pressurized drum that was improperly opened damaged the facility, caused worker contamination, and released radioactive material.

Response(s):

On August 7, 1995, Ecology transmitted a letter to DOE closing this action. This item was

U.S. Department of Energy
Office of River Protection
P.O. Box 450
Richland, Washington 99352

0061562

04-ED-031

APR 05 2004

Mr. Michael A. Wilson, Program Manager
Nuclear Waste Program
State of Washington
Department of Ecology
1315 W. Fourth Avenue
Kennewick, Washington 99336

RECEIVED
APR 14 2004
EDMC

Dear Mr. Wilson:

NEW SOURCE REVIEW EXEMPTION NOTIFICATION FOR THE TANK FARM MIXED WASTE TREATMENT, PACKAGING, AND STORAGE UNIT(S)

Attached for your review and approval is the "New Source Review Exemption Notification for the Tank Farm Mixed Waste Treatment, Packaging, and Storage Unit(s)."

This New Source Review exemption notification is based on emissions of criteria pollutants being below Washington Administrative Code (WAC) 173-400-110(5)((d) threshold levels. Also, emissions of toxic air pollutants are below WAC 173-460 Small Quantity Emission Rates and Acceptable Source Impact Levels.

If you have any questions, please contact me, or your staff may contact Dennis W. Bowser, Environmental Division, (509) 373-2566.

Sincerely,

Roy J. Schepens
Manager

ED:DWB

Attachment

cc: See page 2

cc w/attach:
D. Carrell, CH2M HILL
C. J. Kemp, CH2M HILL
J. Cox, CTUIR
S. Harris, CTUIR
J. L. Hensley, Ecology
S. Lijek, Ecology
J. J. Lyon, Ecology
O. S. Wang, Ecology
W. E. Green, FHI
P. Sobotta, NPT
R. Jim, YN
Administrative Record
CH2M Correspondence Control
Environmental Portal, LMSI

cc w/o attach:
B. G. Erlandson, BNI
M. F. Jarvis, RL
A. W. Conklin, WDOH

Attachment
04-ED-031

New Source Review Exemption Notification for the Tank Farm Mixed Waste Treatment, Packaging, and Storage Unit(s)

NEW SOURCE REVIEW EXEMPTION NOTIFICATION FOR THE TANK FARM MIXED WASTE TREATMENT, PACKAGING, AND STORAGE UNIT(S)

Prepared by
CH2M HILL Hanford Group, Inc.

Date Published
March 2004

United States Department of Energy
Office of River Protection
P.O. Box 450
Richland, Washington

SUMMARY

The U.S. Department of Energy is proposing the installation of one Tank Farm Mixed Waste Treatment, Packaging, and Storage Unit to treat mixed waste from Single-Shell Tanks 241-B-201 through 241-B-204; 241-T-201 through 241-T-204; and 241-T-111. These nine tanks are located in the 200 East and 200 West Areas of the Hanford Site.

This document evaluates the incoming waste stream from the nine single-shell tanks as identified in Tank Waste Information Network System and finds that emissions of organic compounds from treatment of the waste stream would be below Washington Administrative Code regulations for Small Quantity Emission Rates and Acceptable Source Impact Levels would not be exceeded. Dispersion calculations of annual concentration of toxic air pollutants indicate that concentrations of these compounds at the site boundary are within regulatory threshold limits.

In addition, this document evaluates the potential emissions of particulate matter and particulate matter under 10 microns from proposed operations of the Tank Farm Mixed Waste Treatment, Packaging, and Storage Unit. Calculation results show emissions of particulate matter and particulate matter under 10 microns are within regulatory threshold limits.

IN ACCORDANCE WITH WAC 173-400-110(5), A NEW SOURCE REVIEW EXEMPTION IS BEING REQUESTED.

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APPENDICES

APPENDIX A — MAXIMUM REPORTED POLLUTANT – 241-B TANK FARM (TANKS 241-B-201 THROUGH 241-B-204)

APPENDIX B — MAXIMUM REPORTED POLLUTANT – 241-T TANK FARM (TANKS 241-T-201 THROUGH 241-T-204 AND 241-T-111)

APPENDIX C — EMISSION CALCULATION – ACTIVE VENTILATION 241-B TANK FARM (TANKS 241-B-201 THROUGH 241-B-204), 241-T TANK FARM (TANK 241-T-201 THROUGH 241-T-204) AND 241-T-111

APPENDIX D — EMISSION CALCULATION – PARTICULATES

FIGURES

1. Hanford Site 4
2. Location of 241-B Single-Shell Tank Farm within 200E Area 5
3. Location of 241-T Single-Shell Tank Farm within 200W Area 6
4. 241-B Tank Farm Unit Layout 9
5. 241-T Tank Farm Unit Layout 10

TABLES

1. Single-Shell Tanks Covered by the Notice of Construction 1

TERMS

ANSI	American National Standards Institute
ASIL	Acceptable Source Impact Level
ASME	American Society of Mechanical Engineers
°C	degrees Celsius
CAS	Chemical Abstract Services
Cfm	cubic feet per minute
cm	centimeters
m^3/min	cubic meters per minute
CWC	Central Waste Complex
DOE	U.S. Department of Energy
EPDM	ethylene propylenediene monomer (gasket)
ETF	Effluent Treatment Facility
°F	degrees Fahrenheit
gal	gallons
HEPA	high-efficiency particulate air (filter)
HIHTL	hose-in-hose transfer lines
HPS	Health Physics Society
hrs	hours
hrs/yr	hours per year
in.	inches
L	liter
lbs/hr	pounds per hour
lbs/yr	pounds per year
$\mu g/m^3$	micrograms per cubic meter
mg/m^3	milligrams/cubic meter
NSR	New Source Review
OGTS	Offgas Treatment System
PM	particulate matter
PM_{10}	particulate matter under 10 microns
SQER	Small Quantity Emission Rate
SST	single-shell tank
TAP	toxic air pollutant
TWINS	Tank Waste Information Network System
Unit	Hanford Tank Farm Mixed Waste Treatment, Packaging, and Storage Unit
WAC	*Washington Administrative Code*
WRS	waste retrieval system

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

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

1.0 INTRODUCTION

This document serves as the New Source Review (NSR) exemption notification for the installation and operation of one Hanford Tank Farm Mixed Waste Treatment, Packaging, and Storage Unit (Unit) for processing of mixed waste from nine Single-Shell Tanks (SSTs) located in the 241-B and 241-T Tank Farms on the Hanford Site. This exemption notification is submitted in accordance with Washington Administrative Code (WAC) 173-400-110 and is based on emissions from the waste packaging operations being below WAC 173-400-110(5)(d) threshold limits. The scope of this NSR exemption notification begins at the interface point between the Unit and waste retrieval operations and continues through the mixed waste treatment and packaging process to waste container storage. Table 1 lists the nine SSTs that are the subject of this notification.

Table 1. Single-Shell Tanks Covered by the Notice of Construction

Tank	Location	Geodetic Coordinates[1]		Waste Volume*	
		Latitude	Longitude	Liters (L)	Gallons (Gal)
241-B-201	200 East, 241-B Tank Farm			114,000	30,000
241-B-202	200 East, 241-B Tank Farm			110,000	29,000
241-B-203	200 East, 241-B Tank Farm			197,000	52,000
241-B-204	200 East, 241-B Tank Farm			193,000	51,000
241-T-111	200 West, 241-T Tank Farm			1,694,000	447,000
241-T-201	200 West, 241-T Tank Farm			117,000	31,000
241-T-202	200 West, 241-T Tank Farm			80,000	21,000
241-T-203	200 West, 241-T Tank Farm			140,000	37,000
241-T-204	200 West, 241-T Tank Farm			140,000	37,000
				TOTAL 2,785,000 (approx)	TOTAL 735,000 (approx)

*Source: HNF-EP-0182, *Waste Tank Summary Report for Month Ending, May 31, 2003*, CH2M HILL Hanford Group, Inc., Richland, Washington.

[1]Specific Geodetic Coordinates of Hanford tank farm locations have been purposely omitted for security purposes. Approximate locations for the tanks subject to this NSR Exemption Notification can be seen on Figures 2 and 3.

Mixed liquid waste streams will be received from SSTs 241-B-201, -202, -203, -204 (241-B-200 Series Tanks); and 241-T-201, -202, -203, -204 (241-T-200 Series Tanks); and Tank 241-T-111. The 241-B-200 series tanks are 208,500 Liters (L) (55,000 gallons [gal]) capacity tanks. The 241-T-200 series tanks are 208,500 L (55,000 gal) capacity tanks. Tank 241-T-111 is a 2,009,000 L (530,000 gal) capacity tank. All nine SSTs were constructed in 1944.

The Unit will receive approximately 2,785,000 L (735,000 gal) of liquid mixed waste from these nine SSTs.

The owner/operators intend to maximize, to the extent practicable, the storage of packaged waste at existing permitted facilities on the Hanford site, e.g., Central Waste Complex (CWC). If adequate storage is not available at an existing permitted facility when required, packaged waste will be stored at the two container storage areas. This NSR exemption notification includes the construction of two container storage areas. After the liquid waste stream is processed, liquid effluent will be temporarily stored in two liquid effluent holding tankers until transferred to the Effluent Treatment Facility (ETF) for treatment and disposal.

2.0 SCOPE

The scope of this document is to analyze the organic pollutant vapor data from each of the nine tanks as reported in the Tank Waste Information Network System (TWINS) and to calculate, using the maximum reported value of a toxic air pollutant (TAP), an offsite concentration based on a stack discharge flow rate assuming continuous operation of 8,760 hours (hrs) per year. The calculated results are then compared to WAC 173-400 and 173-460 regulations for Small Quantity Emission Rate (SQER) and Acceptable Source Impact Level (ASIL) thresholds. Also, this document analyzes the potential particulate matter emissions of particulate matter (PM) and particulate matter less than 10 microns (PM_{10}) from sorbent material and sand handling. Calculations show PM and PM_{10} are within regulatory threshold limits.

Results of the analysis presented herein support the NSR exemption notification from new source review.

3.0 BACKGROUND

The 241-B-200 series tanks, the 241-T-200 series tanks, and Tank 241-T-111 were constructed in 1944 as part of the Manhattan Project.

Tanks 241-B-201 through 241-B-204 are cascading tanks that received waste from the plutonium concentration activities conducted from October 1946 through June 1952 in the 224-B Concentration Building. After spent nuclear fuel processing activities ended in the 224-B-Plant and cleanout of the plant inventory, Tanks 241-B-201 through 241-B-204 received flush solutions from equipment cleanout in the 221-B and 224-B buildings and metal waste transfer lines (RPP-13300).

Tank 241-T-201 received waste from plutonium concentration activities conducted from November 1946 through May 1949 in the 224-T Concentration Building. Tanks 241-T-202 through 241-T-204 are cascading tanks and received wastes from plutonium concentration activities conducted from May 1949 through May 1952 in the 224-T Concentration Building (RPP-13300).

Tank 241-T-111 received waste from the second decontamination cycle from reprocessing of spent nuclear fuel at the 221-T-Plant from January 1945 through October 1956, cell drainage waste from June 1951 through October 1956, 224-T Building wastes from May 1952 through October 1956, and equipment decontamination waste from February 1960 through June 1967 (RPP-13873).

All nine SSTs have since been interim stabilized. The 241-B-200 series tanks were interim stabilized between 1981 and 1985; the 241-T-200 series tanks in 1981; and Tank 241-T-111 in 1995 (HNF-EP-0182).

4.0 LOCATION

Tanks 241-B-201, 241-B-202, 241-B-203, and 241-B-204 are located on the U.S. Department of Energy (DOE) Hanford Site, 241-B Tank Farm, 200 East Area, Richland, Washington. Tanks 241-T-201, 241-T-202, 241-T-203, 241-T-204, and 241-T-111 are located in the 241-T Tank Farm in the 200 West Area of the Hanford Site (Figures 1, 2, 3).

The facility is managed and operated by CH2M HILL Hanford Group, Inc., for the DOE Office of River Protection under contract DE-AC27-99RL-14047.

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5.0 WASTE PACKAGING SYSTEM

5.1 SYSTEM LAYOUT

Figure 4 shows the Unit layout in the 241-B Tank Farm and Figure 5 shows the Unit layout in the 241-T Tank Farm. The Unit will be initially located in the 241-B Tank Farm and then relocated to the 241-T Tank Farm. The receiving tanks are skid-mounted and the liquid effluent holding tanks are fixed-axle tankers. All tanks will have secondary containment and will be monitored for leaks. Piping outside of the containment structures and trailers at the Unit will be hose-in-hose transfer lines (HIHTL). Commercial unit support systems include above ground, double-contained, compressed air, instrument air, chilled water, deionized water, and steam supply. Electrical services will be provided by Hanford Site utilities. Office and field trailers will be set-up to accommodate the field staff. Staging areas will be constructed for incoming empty waste containers. A permitted container storage area will be constructed in the 200 East and 200 West Areas for staging of empty containers and filled waste containers if storage space is not available when required at the existing onsite permitted storage areas.

5.2 WASTE RETRIEVAL

The Unit will receive mixed waste from a SST waste retrieval system (WRS). Waste retrieval activities from the nine SSTs are outside the scope of this NSR exemption notification. Information on waste retrieval is provided solely as information to describe the interfaces between the WRS activities and the Unit.

The diluted tank waste will be batch-transferred from the SST WRS using HIHTL to the interface point at the Unit. A leak proof connection between the HIHTL and the Unit will be established. The Unit will have a capacity to treat 55,510 L (14,400 gal) of diluted tank waste per day. Tank waste will be transferred to the Unit in batches.

5.3 WASTE FEED RECEIPT TANKS

After the mixed waste stream is retrieved from the nine SSTs identified in this NSR exemption notification, the waste stream enters the Unit for placement into four waste feed receipt tanks. A fifth receipt tank is typically maintained with no inventory to remain available for contingency use. The waste feed receipt tanks are designed to receive and contain the waste transferred by the WRS activities in preparation for treatment by the Unit.

The waste feed receipt tanks performs the following functions:

- Receive the daily diluted tank waste feed
- Homogenize the received waste
- Feed the mixer dryer.

Each waste feed receipt tank is skid-mounted and will have secondary containment with leak detection capability. Interconnecting piping will be made of material compatible with the waste stream.

The waste feed receipt tanks are designed to accommodate the total amount of waste transferred per batch. Each tank has a capacity of 30,283 L (8,000 gal) with a net working volume of 27,255 L (7,200 gal). Control valves will regulate the incoming waste flow to the first set of receiving tanks. Once the liquid waste is received from retrieval operations, the waste will be transferred from the waste receipt tanks to a mixer dryer.

Waste feed receipt tanks will be actively ventilated via an Offgas Treatment System (OGTS) (Section 5.8). Waste receipt tanks will be equipped with high-efficiency particulate air (HEPA) inlet breather vents. During Unit operation, the waste feed receipt tanks are under negative pressure and offgases will be connected to the OGTS. Outside air will enter the waste feed receipt tanks through the HEPA filtered inlet breather vent.

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5.4 MIXER DRYER

The tank waste is transferred from the waste feed receipt tanks to a mixer dryer that evaporates liquids prior to packaging.

Two mixer dryers will be operated to separate liquids from the solids such that the resultant solid waste product will have residual water of approximately 20% weight by volume. As diluted tank waste enters the mixer dryer, pressure will be reduced (i.e., a vacuum is applied) and the temperature will be adjusted to enhance the evaporation rate. Temperature will be increased up to 60 degrees Celsius (°C) (140 degrees Fahrenheit [°F]) using a packaged boiler unit. The mixer dryer will utilize heat from the steam through a heat exchanger for drying the waste. The power source for the packaged boiler unit will be from Hanford's onsite electrical power grid. A back-up generator power source is not anticipated. Liquid condensate effluent from the drying operation will be routed to the liquid effluent holding tanks. When the drying cycle is complete, the waste will be mixed with sorbent from the Bulk Material Handling System and readied for waste packaging.

5.5 WASTE PACKAGING

The primary function of waste packaging is to fill and sample the waste container, decontaminate (if necessary) the waste container, and release the filled containers to a permitted container storage area. The waste packaging process will be enclosed within a containment structure measuring approximately 3.7 m (12 ft) wide, 3.7 m (12 ft) high, and 12.2 m (40 ft) long and operates under negative pressure via the OGTS.

The waste packaging process within the containment structure will consist of three isolated zones that will be separated by roll-up doors acting as air barriers to prevent the spread of contamination. Each isolation area will be maintained under negative air pressure to prevent potentially contaminated air from escaping to the atmosphere. Negative pressure will be maintained as airflow moves from areas of least contamination to areas of greater contamination. The highest negative pressure will be maintained in the waste container packaging area. Then, a decreasing negative pressure will be maintained moving away from the packaging station to the airlock areas.

An empty waste container will be loaded into the load-in station by a forklift. The container will be inspected and then the lid will be removed at the lidding station and the container (with pre-installed waste bag) will be positioned under an enclosure for filling. The enclosure also has a separate connection to the OGTS to maintain negative pressure in addition to the lidding station isolation zone.

Dewatered tank waste from the mixer dryer and sorbent will be metered into a container using a control valve. The waste packaging activity will be visually inspected and redundantly monitored using closed-circuit television. The waste bag will then be closed and the container moved to the survey and decontamination station where the lid will be replaced. The container will be sealed using a bolted lid with an ethylene propylenediene monomer (EPDM) gasket measuring approximately 3.81 centimeters (cm) (1.5 inches [in.]) wide and 1.27 cm (0.5 in.) thick. The EPDM gasket will be compatible with and meet the waste packaging acceptance

criteria of the offsite deep geologic repository. NucFil[1] HEPA filters or equivalent will be installed to relieve any potential internal gas generation. Treated waste will be packaged in standard waste boxes. Open-top 208 L (55 gal) steel drums will be used for miscellaneous waste from Unit operations.

Waste containers will move on a conveyance system within the containment structure using local controls to manually position the container. Position sensors and positive mechanical stops will control the container movement on the conveyance system during the packaging process. After filling, waste containers will be surveyed and decontaminated as appropriate prior to moving containers to the container storage area.

5.6 SORBENT ADDITION

Once the required level of dryness is achieved and prior to discharge from the mixer dryer, a solid drying agent (sorbent) material will be added to the dried waste from the Bulk Material Handling System. Sorbent material is bulk loaded into a storage silo above the mixer dryer and discharged into the mixer dryer as needed. The purpose of the sorbent is to remove any free water that may form during container handling and transportation. When needed, sand will be added to scour the mixer dryer drum walls to prevent any buildup of solids that could affect mixer dryer heating operations.

5.7 LIQUID EFFLUENT HOLDING TANK

Liquid effluent condensed from the mixer dryer will be collected in two 66,775 L (17,640 gal) liquid effluent holding tankers. A small portion of liquid waste will be pumped back to the SST WRS as needed and the remaining liquid effluent will be filtered and transferred via a waste transfer tanker truck to the ETF for treatment and disposal. Each liquid effluent holding tank will be interconnected to an inlet HEPA breather vent. Outside air will enter the liquid effluent holding tankers through a HEPA filtered inlet breather vent.

5.8 OFFGAS TREATMENT SYSTEM

The OGTS will receive offgas from the Unit's five waste feed receipt tanks, two mixer dryers, and waste packaging operations within the treatment/packaging trailer. The OGTS is a skid-mounted unit that will filter and monitor the effluents for radioactive particulates and exhaust to the atmosphere through a stack. Airflow entering the OGTS passes through a separator to remove water vapor and a heater to raise the temperature and dew point prior to reaching the HEPA filters. After the airflow is heated, airflow continues through a pre-filter, two HEPA filters in series, and then exhausted to the atmosphere through a stack equipped with sample ports and monitoring equipment.

The OGTS will be fabricated in accordance with latest versions of American Society of Mechanical Engineers (ASME) AG-1, ASME N509, ASME N510, and American National Standards Institute (ANSI) ANSI/Health Physics Society (HPS) N13.1. Each HEPA filter section will have injection ports and sampling ports for each HEPA for independent aerosol testing. The stack will include an effluent monitoring system that will indicate high radiation

[1] NucFil is a registered trademark of Nuclear Filter Technology, Inc., Golden, Colorado

levels by sounding an alarm. If a high radiation alarm sounds, control room operators will follow alarm response procedures that will shut down the process in a safe and controlled manner. Specific steps that operators will take to initiate a controlled shutdown will consist of the following activities:

- Stop steam to dryer (removes motive force)
- Stop waste packaging (allows operators to leave container)
- Stop Liquid Effluent System transfers (allows operators to clear area)
- Stop slurry feed to dryers (precaution only).

The stack monitoring system will include a stack sampling shrouded nozzle design and vacuum pumps to provide sampling ability. Each OGTS section will drain to a central seal pot with a water seal to prevent any bypass of airflow through the drains. The seal pot will be monitored for level to ensure the drain seals are maintained.

The airflow exhaust design velocity is approximately 60 cubic meters per minute (m^3/min) (2,000 cubic feet per minute [cfm]). It is anticipated the exhauster may operate at less than 60 m^3/min (2,000 cfm) but would not exceed this flow rate.

5.9 WASTE STORAGE

The owner/operators intend to maximize, to the extent practicable, the storage of packaged waste at existing permitted facilities on the Hanford site, e.g., Central Waste Complex (CWC). If adequate storage is not available at an existing permitted facility when required, packaged waste will be stored at the two container storage areas. This NSR exemption notification includes the construction of two container storage areas. The 200 East container storage area will be designed to accommodate up to 240 filled waste containers and the 200 West container storage area is will be designed to accommodate up to 1260 filled waste containers.

6.0 PROJECT ACTIVITIES

The proposed activity consists of construction, installation, and operation on the Hanford Site of one Unit. This Unit will dewater and dry (i.e., treat) a mixed waste stream, add sorbent material, package the dried mixed waste stream, and store the packaged containers. The Unit will be constructed and operated to receive and package waste streams from Hanford Site waste retrieval activities from nine SSTs as described in Section 1.0 (see also Table 1). Four SSTs are located in the 200 East Area (241-B Tank Farm) and five SSTs are located in the 200 West Area (241-T Tank Farm) of the Hanford Site (Figure 1). The Unit will be set up and operated at the 241-B Tank Farm in the 200 East Area (Figure 2). Upon completion of the activities at the 241-B Tank Farm, the Unit will be relocated to the 241-T Tank Farm (Figure 3). The Unit will treat and package about 2,785,000 L (735,000 gal) of liquid mixed waste from the 241-B and 241-T Tank Farms. SST tank waste will be dewatered and dried to lower the moisture content by utilizing a mixer dryer. After treatment, the waste will be packaged for onsite storage pending shipment to an offsite deep geological repository.

Activities for operating the Unit are expected to include those noted below, and/or others similar in nature. The listing is not intended to be all-inclusive but to only represent those types of activities required in the operation of the Unit. Other Unit activities not yet defined may be conducted with no anticipated impact on the organic inventories of the nine SSTs used for calculating emissions. The noted processes apply to the Unit:

- Receipt of diluted tank waste from waste retrieval activities to waste tanks within the Unit
- Mixing of diluted tank waste in the waste feed receipt tanks
- Transfer of diluted tank waste to the mixer dryer
- Evaporating liquid using a mixer dryer
- Bulk loading and mixing of sorbent
- Collection of dewatered waste/sorbent in waste containers
- Collection of condensate
- Transfer of residual liquid effluent to holding tanks
- Occasional addition of sand to scour mixer dryer walls
- Routing of receiving tank off-gas, mixer dryer offgas, and offgas from waste packaging to an OGTS
- Interim storage of empty waste containers
- Storage of filled waste containers
- Site preparation work – grading, excavating, backfilling, utilities, fencing, storage pads
- Installation of construction office trailers, field trailers, change trailers, storage facilities.

7.0 VENTILATION AND AIRBORNE EMISSIONS CONTROL SYSTEM

Waste feed receipt tanks, mixer dryers, and waste packaging operations will be exhausted under negative pressure to the OGTS. Airflow will enter the OGTS and will pass through a moisture separator to remove water vapor and through a heater to raise the temperature and dew point prior to reaching the HEPA filters. After the heater, airflow will continue through a pre-filter, two HEPA filters, in series, and will then be exhausted to the atmosphere through a 10-in. diameter stack equipped with sample ports and monitoring equipment for radioactive air emissions. Airflow will be exhausted at approximately 60 m^3/min (2,000 cfm). The exhauster may be operated at a flow rate less than 60 m^3/min (2,000 cfm).

To minimize the spread of contaminants during waste packaging activities, three zones of progressively negative pressure will be established within the treatment/packaging trailer. Negative pressure will be maintained as airflow moves from areas of least contamination to areas of greater contamination. The highest negative pressure will be maintained in the waste container packaging area. Then, a decreasing negative pressure will be maintained moving away from the waste packaging area to the airlock areas. Interlocks will be provided to assure zoned areas are maintained under negative pressure during operations. Ventilation system components

will meet the latest requirements of ASME N509, ASME N510, and ASME AG-1, as applicable. Where conflicts occur, ASME AG-1 takes precedence, unless an exemption has been documented in *General WAC 246-247 Technology Standards Exemption Justifications for Waste Tank Ventilation Systems (RPP-19233).* Two Liquid Effluent Holding Tankers will be passively vented directly to atmosphere through a HEPA breather filter. Filter design and performance specifications are the same as those for the OGTS. Packaged waste containers will be equipped with NucFil HEPA filters (or equivalent).

No sampling will be required for nonradioactive air emissions because all contaminant emissions are below their respective SQERs or below their ASILs. Organic vapor analyzers, or other similar instruments for detecting fugitive organic emissions as part of Hanford's Industrial Hygiene program to monitor worker exposure, will be used to monitor for volatile organic compounds.

8.0 METHODOLOGY OF COMPARISON CALCULATED EMISSIONS TO SQER AND ASIL THRESHOLDS

For calculation of dispersed offsite concentration of TAPs to members of the public, vapor sample data as reported in TWINS were used for developing the pollutant source term. Tank vapor space sample data, as available, were reviewed against WAC 173-460-160 Class B TAP Chemical Abstract Services (CAS) numbers and ASILs. The maximum reported concentration was selected for the Class B TAP CAS number for those tanks located in the 241-B Tank Farm (241-B-201 through 241-B-204 [see Appendix A]). Similarly, the maximum reported concentration was selected for the Class B TAP CAS number and for Tanks 241-T-201 through 241-T-204; and 241-T-111 (see Appendix B). A worst-case bounding tank pollutant source inventory was developed that contained the highest concentration of each constituent found. Emission quantities were calculated based on the worst-case tank scenario using maximum concentrations and Unit operation under active ventilation for one year (8,760 hrs). Active ventilation calculations appear in Appendix C and represent a conservative approach.

Appendix D calculations reflect the estimated particulate matter emissions of PM and PM_{10} from sorbent hopper loading.

Assumptions made in calculating emission quantities:

1. A maximum concentration of an organic constituent was selected from tank groupings for the 241-B-200 series; the 241-T-200 series; and Tank 241-T-111 for a worst-case inventory.

Basis: Selecting the maximum value of each identified organic constituent from vapor space data would provide a worst-case tank scenario. Emission calculations would, therefore, be based on a worst-case concentration and would provide a bounding calculation of emissions to the environment.

2. Vapor space data from Tank 241-B-202 as reported in TWINS is used for Tanks 241-B-201 through 241-B-204.

Basis: Vapor space data is only available for Tank 241-B-202, and is not available for Tanks 241-B-201, 241-B-203, and 241-B-204. All of the 241-B-200 series tanks received waste from the 224-B Concentration Building (RPP-13300). Because of the source of the waste from the 224-B Concentration Building and the designed cascading flow effect of the tanks, it is reasonable to assume that the vapor space data in Tanks 241-B-201, 241-B-203, and 241-B-204 would be similar to Tank 241-B-202.

3. Vapor space data from Tank 241-B-202 as reported in TWINS is used for Tanks 241-T-201 through 241-T-204.

Basis: Vapor space data is only available for Tank 241-B-202, and is not available for Tanks 241-T-201 through 241-T-204. The plutonium concentration process conducted in the 224-T Concentration Building was similar to that conducted at the 224-B Concentration Building. Both facilities operated under the same flowsheet processing conditions. No other waste types were transferred into these tanks. Since Tank 241-B-202 is the only 241-B-200 series tank with vapor space data, it is reasonable to assume that the vapor space data for 241-B-202 would be representative of the vapor space data for Tanks 241-T-201 through 241-T-204 because the same plutonium concentration process was conducted at both the 224-B and 224-T Concentration Building. Therefore, the 241-T-200 series tanks have been considered as having a similar amount of ammonia present in the vapor space as reported in TWINS for Tank 241-B-202.

4. The exhauster flow rate is 60 m^3/min (2,000 cfm)

Basis: The flow rate is the current design basis.

5. The OGTS will operate 24 hrs/day for 365 days/yr for a total of 8,760 hrs.

Basis: This is very conservative but does represent the maximum run time for emission calculations.

6. The Unit Concentration Factors from Industrial Source Complex Dispersion Modeling were used for ground level releases for 24-hr average and annual average releases from the 200 East and 200 West Areas. 24-hr average concentration factor = 2.79 (200 East) and 3.46 (200 West) $\mu g/m^3/g/sec$; annual average = 0.0793 (200 East) and 0.0585 (200 West) $\mu g/m^3/g/sec$.

Basis: These unit concentration factors are used by the Tank Farm Contractor on the Hanford Site for air concentrations at site boundaries. Source: Memo to J.S. Hill from P.D. Rittman, Unit Concentration Factors from ISC3, September 27, 1996.

The Appendix C worksheet shows the comparison of calculated concentration of TAPs to SQER and ASIL thresholds. Columns A through E list the chemical name, CAS number, TAP Class, SQER, and ASIL respectively, and are from WAC 173-460-080, -150, -160. Column F is the maximum reported value for the CAS number from column H in milligram (mg)/cubic meter (m^3) of Appendix A (for 241-B Tank Farm) or Appendix B (for 241-T Tank Farm). Column G is the calculated result for pounds per hour discharged. Column H is the result for pounds/year (lbs/yr) (8,760 hours/year [hrs/yr] * pounds per hour [lbs/hr] from Column G). Column I is the result of a comparison of Column G lbs/hr or Column H lbs/yr to the Column D SQER value noting either the calculated result is at or below the SQER. Column J is the result of a

conversion from Column H lbs/yr to $\mu g/m^3$. Column K is the result of a comparison between Column J and Column E noting if the calculated value is below the ASIL threshold.

Organic chemicals in the waste stream received from the nine SSTs and treated at the Unit contain the same concentrations as reported in TWINS and are expected to be in the vapor space of the waste feed receipt tanks. Estimated emissions are conservative because calculated emissions were assumed to occur over an 8,760 hr operating period and no credit was taken for the purge time of the receiving tanks under active ventilation. When consideration is given for tank purging, the pollutants will be exhausted to the environment in minutes and not over the period of 8,760 hours.

For illustrative purposes, assume batch transfers occur and there is 90% available vapor space area for each of the four waste receipt tanks totaling 98,120 L (25,920 gal). At a designed exhaust ventilation rate of 60 m^3/min (2,000 cfm), the total vapor space volume would be purged in approximately 0.03 hours, or approximately 2 minutes.

2,000 cfm*7.481 gal/cu ft*60 min/hr = 8.98E+05 gal/hr

25,920 gal/8.98E+05 gal/hr = 0.03 hrs or about 2 minutes

Therefore, when batch transfers of waste are received at the waste receipt tanks and purged within minutes, estimating emissions of TAPs over a total period of 8,760 hours is conservative, yet still below SQER and ASIL thresholds.

Appendix D is the calculation and results of particulate emissions from loading of the storage silo with sorbent and sand. Results show PM and PM_{10} emissions are well within regulatory threshold limits of WAC 173-400-110(5)(d).

9.0 REFERENCES

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RPP-19233, 2004, *General WAC 246-247 Technology Standards Exemption Justifications for Waste Tank Ventilation Systems*, Rev. 0, CH2M HILL Hanford Group, Inc., Richland, Washington.

TWINS, *Tank Waste Information Network System*, available on the Internet to authorized personnel at http://twinsweb.pnl.gov, as of August 2003.

WAC 173-400, "General Regulations for Air Pollution Sources," *Washington Administration Code*, as amended.

WAC 173-460, "Controls for New Sources of Toxic Air Pollutants," *Washington Administrative Code*, as amended.

APPENDIX A

MAXIMUM REPORTED POLLUTANT – 241-B TANK FARM (TANKS 241-B-201 THROUGH 241-B-204)

Table A-1. Maximum Reported TAP for 241-B Tank Farm

Column A	B	C	D	E	F	G	H	I	J	K	L
Toxic Air Pollutant	Molecular Weight	CAS #	TAP CLASS	SQER (A lbs/yr, B lbs/hr)	ASIL ($\mu g/m^3$)	Maximum Reported (ppbv)	Maximum Reported (mg/m^3)	B201 ppmv Max Value	B202 ppmv Max Value (a)	B203 ppmv Max Value	B204 ppmv Max Value
							H=G*B/24/1000				
Ammonia	17.0306	7664-41-7	B	2.00	100	3.0E+03	2.1E+00	ND	3.00E+00	ND	ND

Note: (a) Ammonia (reported as 3 ppmv) has been converted to ppbv

Conversions:
3.0 ppmv *1000ppbv/ppmv = 3,000 ppbv
mg/m^3 = ppbv * molecular weight/24/1000 [2.1E+00mg/m^3 = 3.0E+3ppbv*17.0306/24/1000]

Source:Tank Waste Inventory Network System (TWINS), http://twinsweb.pnl.gov
ND - no data available

APPENDIX B

MAXIMUM REPORTED POLLUTANT – 241-T TANK FARM

(TANKS 241-T-201 THROUGH 241-T-204 AND 241-T-111)

Table B-1. Maximum Reported TAP for 241-T Tank Farm

Column A	B	C	D	E	F	G	H	I	J	K	L	M
Toxic Air Pollutant	Molecular Weight	CAS #	TAP CLASS	SQER (A lbs/yr, B lbs/hr)	ASIL ($\mu g/m^3$)	Maximum Reported (ppbv)	Maximum Reported (mg/m^3)	T201 ppmv Max Value	T202 ppmv Max Value	T203 ppmv Max Value	T204 ppmv Max Value	T111 ppmv Max Value (a)
							H=G*B/24/1000					
Chloroform	119.3781	67-66-3	All	10	0.043	1.2E+01	6.0E-02					1.20E+01
Benzene	78.1147	71-43-2	All	20	0.12	6.0E+00	2.0E-02					6.00E+00
Dichloromethane (methylene chloride)	84.933	75-09-2	All	50	0.56	9.0E+00	3.2E-02					9.00E+00
Perchloroethylene (tetrachloroethylene)	165.8342	127-18-4	All	500	1.1	2.1E+01	1.5E-01					2.10E+01
Phenyl ether	170.2129	101-84-8	B	0.20	23	2.8E+01	2.0E-01					2.80E+01
Pyridine	79.1023	110-86-1	B	0.60	53	2.0E+01	6.6E-02					2.00E+01
2-Hexanone (MBK)	100.1619	591-78-6	B	1.20	67	8.0E+00	3.3E-02					8.00E+00
Ammonia	17.0306	7664-41-7	B	2.00	100	2.3E+05	1.6E+02	3.00E+00	3.00E+00	3.00E+00	3.00E+00	2.27E+02
Acetonitrile	41.0529	75-05-8	B	2.60	220	9.4E+01	1.6E-01					9.40E+01
Toluene	92.1419	108-88-3	B	5	400	1.7E+01	6.5E-02					1.70E+01
n-Butyl alcohol	74.1237	71-36-3	B	5	500	7.0E+00	2.2E-02					7.00E+00
Methyl isobutyl ketone (MIBK)	100.1619	108-10-1	B	5	680	4.0E+00	1.7E-02					4.00E+00
Methyl n-amyl ketone	114.189	110-43-0	B	5	780	1.2E+01	5.7E-02					1.20E+01
Methyl alcohol	32.0424	67-56-1	B	5	870	3.0E+02	4.0E-01					3.00E+02
tert-Butyl alcohol	74.1237	75-65-0	B	5	1,000	2.0E+00	6.2E-03					2.00E+00
Methyl ethyl ketone (MEK)	72.1077	78-93-3	B	5	1,000	1.5E+01	4.5E-02					1.50E+01
n-Propyl alcohol	60.0966	71-23-8	B	5	1,600	7.0E+00	1.8E-02					7.00E+00
Methyl propyl ketone	86.1348	107-87-9	B	5	2,300	2.0E+01	7.2E-02					2.00E+01
Isopropyl alcohol	60.0966	67-63-0	B	5	3,300	5.0E+00	1.3E-02					5.00E+00
Nonane	128.2597	111-84-2	B	5	3,500	1.7E+01	9.1E-02					1.70E+01
Octane	114.2326	111-65-9	B	5	4,700	1.0E+00	4.8E-03					1.00E+00
Heptane (n-Heptane)	100.2055	142-82-5	B	5	5,500	2.0E+00	8.4E-03					2.00E+00
Acetone	58.0806	67-64-1	B	5	5,900	1.9E+02	4.5E-01					1.87E+02
Trichlorofluoromethane	137.3685	75-69-4	B	5	19,000	1.8E+01	1.0E-01					1.80E+01

Ammonia Conversions:
Note: (a) The maximum Ammonia value (reported as 2.27E+02 ppmv) for T-111 has been converted to 2.3E+05 ppbv (Column G)

2.27E+02 ppmv *1000ppbv/ppmv = 227,000 ppbv

mg/m^3 = ppbv * molecular weight/24/1000 [1.60E+02mg/m^3=2.3E+05ppbv*17.0306/24/1000]

Tank Waste Inventory Network System (TWINS), http://twinsweb.pnl.gov

Note: The 241-T-200 Series tank have been considered as having a similar amount of ammonia present in the vapor space as reported in TWINS for Tank 241-B-202.

APPENDIX C

EMISSION CALCULATION – ACTIVE VENTILATION 241-B TANK FARM (241-B-201 THROUGH 241-B-204) 241-T TANK FARM (241-T-201 THROUGH 241-T-204 AND 241-T-111)

Table C-1. 241-B Tank Farm – Active Ventilation

VENTILATION RATE	2000	Cfm
HOURS PER YEAR OPERATION	8,760	HOURS
TAP A CONCENTRATION FACTOR (ACF)	0.0793	(200-EAST)
TAP B CONCENTRATION FACTOR (BCF)	2.79	(200-EAST)

Column A	B	C	D	E	F	G	H	I	J	K
Toxic Air Pollutant	CAS #	TAP CLASS	SQER (A LBS/YR, B LBS/HR)	ASIL ($\mu g/m^3$)	Maximum Reported (mg/m^3)	LBS/HR DISCHARGED	LBS/YR DISCHARGED	AT OR BELOW SQER	ESTIMATED OFFSITE CONCENTRATION ($\mu g/m^3$)	BELOW ASIL VALUE
Ammonia	7664-41-7	B	2	1.00E+02	2.1E+00	1.57E-02	1.38E+02	YES	5.53E-03	YES

EXAMPLE CALCULATIONS:

TAP B - AMMONIA G= 2000*2.1E+00*(1/1000)*(1/35.3147)*(1/453.59237)*60
H= 8,760*1.57E-02
J= 1.57E-02*24*453.59237*2.79/86400

Table C-2. 241-T Tank Farm – Active Ventilation

VENTILATION RATE	2000	Cfm
HOURS PER YEAR OPERATION	8,760	HOURS
TAP A CONCENTRATION FACTOR (ACF)	0.0585	(200-WEST)
TAP B CONCENTRATION FACTOR (BCF)	3.46	(200-WEST)

A	B	C	D	E	F	G	H	I	J	K
Toxic Air Pollutant	CAS #	TAP CLASS	SQER (A LBS/YR, B LBS/HR)	ASIL ($\mu g/m^3$)	Maximum Reported (mg/m^3)	LBS/HR DISCHARGED	LBS/YR DISCHARGED	AT OR BELOW SQER	ESTIMATED OFFSITE CONCENTRATION ($\mu g/m3$)	BELOW ASIL VALUE
Chloroform	67-66-3	All	10	4.30E-02	6.0E-02	4.51E-04	3.95E+00	YES	3.32E-06	YES
Benzene	71-43-2	All	20	1.20E-01	2.0E-02	1.50E-04	1.31E+00	YES	1.10E-06	YES
Dichloromethane (methylene chloride)	75-09-2	All	50	5.60E-01	3.2E-02	2.40E-04	2.10E+00	YES	1.77E-06	YES
Perchloroethylene (tetrachloroethylene)	127-18-4	All	500	1.10E+00	1.5E-01	1.12E-03	9.84E+00	YES	8.28E-06	YES
Phenyl ether	101-84-8	B	0.20	2.30E+01	2.0E-01	1.50E-03	1.31E+01	YES	6.53E-04	YES
Pyridine	110-86-1	B	0.80	5.30E+01	6.6E-02	4.94E-04	4.33E+00	YES	2.16E-04	YES
2-Hexanone (MBK)	591-78-6	B	1.20	6.70E+01	3.3E-02	2.47E-04	2.17E+00	YES	1.08E-04	YES
Ammonia	7664-41-7	B	2	1.00E+02	1.6E+02	1.20E+00	1.05E+04	YES	5.23E-01	YES
Acetonitrile	75-05-8	B	3	2.20E+02	1.6E-01	1.20E-03	1.05E+01	YES	5.23E-04	YES
Toluene	108-88-3	B	5	4.00E+02	6.5E-02	4.87E-04	4.27E+00	YES	2.12E-04	YES
n-Butyl alcohol	71-36-3	B	5	5.00E+02	2.2E-02	1.65E-04	1.44E+00	YES	7.18E-05	YES
Methyl isobutyl ketone (MIBK)	108-10-1	B	5	6.80E+02	1.7E-02	1.27E-04	1.12E+00	YES	5.55E-05	YES
Methyl n-amyl ketone	110-43-0	B	5	7.80E+02	5.7E-02	4.27E-04	3.74E+00	YES	1.86E-04	YES
Methyl alcohol	67-56-1	B	5	8.70E+02	4.0E-01	3.00E-03	2.62E+01	YES	1.31E-03	YES
tert-Butyl alcohol	75-65-0	B	5	1.00E+03	6.2E-03	4.64E-05	4.07E-01	YES	2.02E-05	YES
Methyl ethyl ketone (MEK)	78-93-3	B	5	1.00E+03	4.5E-02	3.37E-04	2.95E+00	YES	1.47E-04	YES
n-Propyl alcohol	71-23-8	B	5	1.60E+03	1.8E-02	1.35E-04	1.18E+00	YES	5.88E-05	YES
Methyl propyl ketone	107-87-9	B	5	2.30E+03	7.2E-02	5.39E-04	4.72E+00	YES	2.35E-04	YES
Isopropyl alcohol	67-63-0	B	5	3.30E+03	1.3E-02	9.74E-05	8.53E-01	YES	4.25E-05	YES
Nonane	111-84-2	B	5	3.50E+03	9.10E-02	6.82E-04	5.97E+00	YES	2.97E-04	YES
Octane	111-65-9	B	5	4.70E+03	4.80E-03	3.60E-05	3.15E-01	YES	1.57E-05	YES
Heptane (n-Heptane)	142-82-5	B	5	5.50E+03	8.40E-03	6.29E-05	5.51E-01	YES	2.74E-05	YES
Acetone	67-64-1	B	5	5.90E+03	4.50E-01	3.37E-03	2.95E+01	YES	1.47E-03	YES
Trichlorofluoromethane	75-69-4	B	5	1.90E+04	1.00E-01	7.49E-04	6.56E+00	YES	3.27E-04	YES

EXAMPLE CALCULATIONS:

TAP A - BENZENE	G= 2000*2.0E-02*(1/1000)*(1/35.3147)*(1/453.59237)*60
TAP A & B	H= G*8,760
TAP A - BENZENE	J= 1.31E+00*453.59237*.0585/31536000
TAP B - AMMONIA	J= 1.20E+00*24*453.59237*3.46/86400

APPENDIX D

EMISSION CALCULATION – PARTICULATES

Table D-1. Particulate Matter/Particulate Matter Under 10 Microns Emissions

SORBENT HOPPER LOADING									
PM					0.0035	lb/1000 lb material added			
PM_{10}					0.0014	lb/1000 lb material added			
Operating Schedule									
24hrs/day*7 days/week*52weeks/year					8760	Hours/year			
Equipment Availability Factor (Annual)					100%				
Control Equipment provided is cyclone. Correction for collection efficiency is:					1.013	PM			
	NOTE: EPA Publication AP-42, Section 11.26 (Talc Processing)				1.04	PM_{10}			
Sorbent Addition Rate (lb/hr)	Bag Weight (lb)	Interval Between Bag Loading (hrs)	Bag Loading Duration (min)	Number of Bags Loaded Per Year		Uncorrected Emission Rates (PM)	Uncorrected Emission Rates (PM_{10})	Corrected Emission Rates (PM)	Corrected Emission Rates (PM_{10})
17	1000	59	5	149					
					Hourly (lb/hr)	3.50E-03	7.00E-04	3.55E-03	7.28E-04
					Annual (tons/yr)	5.21E-04	5.21E-05	5.28E-04	5.42E-05
	NUMBER OF UNITS	**2**		**TOTAL**	**Hourly (lb/hr)**	**7.00E-03**	**1.40E-03**	**7.09E-03**	**1.46E-03**
				TOTAL	**Annual (tons/yr)**	**1.04E-03**	**1.04E-04**	**1.06E-03**	**1.08E-04**

ENGINEERING CHANGE NOTICE

1. ECN 6346

Page 1 of 3

2. ECN Category (Mark one)	3. Originator's Name, Organization, MSIN, and Telephone No.	4. USQ Required?	5. Date
Supplemental [] Direct Revision [X] Change ECN [] Temporary [] Standby [] Supersedure [] Cancel/Void []	C. A. Petersen 73510 H5-27	[] Yes [X] No	9/18/96
	6. Project Title/No./Work Order No. Storage and Disposal Project	7. Bldg./Sys./Fac. No. -- NA	8. Approval Designa NA
	9. Document Numbers Changed by this ECN (includes sheet no. and rev.) WHC-SD-WM-TI-599, Rev. 1	10. Related ECN No(s). --	11. Related PO No.

12a. Modification Work	12b. Work Package No.	12c. Modification Work Complete	12d. Restored to Original Condition (Temp. or Standby ECN only)
[] Yes (fill out Blk. 12b) [X] No (NA Blks. 12b, 12c, 12d)	-- NA	-- NA Design Authority/Cog. Engineer Signature & Date	-- NA Design Authority/Cog. Engineer Signature & Date

13a. Description of Change

13b. Design Baseline Document? [] Yes [X] No

Document update and add Appendix D.

14a. Justification (mark one)

Criteria Change	[]	Design Improvement	[]	Environmental	[]	Facility Deactivation	[]
As-Found	[X]	Facilitate Const	[]	Const. Error/Omission	[]	Design Error/Omission	[]

14b. Justification Details

Appendix D added to show comparison of 10 CFR 61 and 5820.2a performance requirements.

15. Distribution (include name, MSIN, and no. of copies)

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Technical Basis for Classification of Low-Activity Waste Fraction from Hanford Site Tanks

C. A. Petersen
Westinghouse Hanford Company, Richland, WA 99352
U.S. Department of Energy Contract DE-AC06-87RL10930

EDT/ECN: 617630 UC: 721
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Key Words: low-activity waste, classification, U.S. Nuclear Regulatory Commission, incidental waste

Abstract: The overall objective of this report is to provide a technical basis to support a U.S. Nuclear Regulatory Commission determination to classify the low-activity waste from the Hanford Site single-shell and double-shell tanks as "incidental" wastes after removal of additional radionuclides and immobilization. The proposed processing method, in addition to the previous radionuclide removal efforts, will remove the largest practical amount of total site radioactivity, attributable to high-level waste, for disposal in a deep geologic repository. The remainder of the waste would be considered "incidental" waste and could be disposed onsite.

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M. M. McCarthy		X			

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Table D-2. Performance Requirements Comparisons.

Section 61.52(4) Wastes must be emplaced in a manner that maintains the package integrity during emplacement, minimizes the void spaces between packages, and permits the void spaces to be filled.	Section 3.i.9.d Waste placement into disposal units should minimize voids between containers.
Section 61.52(7) Boundaries and locations of each disposal unit must be accurately located and mapped....	Section 3.i.9.b Permanent identification marks for disposal excavations and monitoring wells shall be emplaced.
Section 61.52(10) Active waste disposal operations must not have an adverse effect on completed closure and stabilization measures.	Section 3.i.9.e Operations are to be conducted so that active waste disposal operations will not have an adverse effect on filled disposal units.
Section 61.53(b) The licensee must have plans for taking corrective measures if migration of radionuclides would indicate that the performance objectives of subpart C may not be met.	Section 3.k.2 The environmental monitoring program shall be designed to measure: (a) operational effluent releases; (b) migration of radionuclides; (c) disposal unit subsidence; and (d) changes in disposal facility and disposal site parameters which may affect long term site performance.
Section 61.53(d) ...the monitoring system must be capable of providing early warning of releases of radionuclides from the disposal site before they leave the site boundary.	Section 3.k.4 The monitoring program shall be capable of detecting changing trends in performance sufficiently in advance to allow application of necessary corrective actions prior to exceeding performance objectives....

Table D-2. Performance Requirements Comparisons.

Section 61.56(a)(6) Waste must not be pyrophoric. Pyrophoric materials contained in waste shall be treated, prepared, and packaged to be nonflammable.	Section 3.i.5.f (Wording identical)
Section 61.56(a)(7) Waste in a gaseous form must be packaged at a pressure that does not exceed 1.5 atmosphere at 20 degrees C. Total activity must not exceed 100 curies per container.	Section 3.i.5.e Waste in a gaseous form must be packaged at a pressure that does not exceed 1.5 atmosphere at 20°C.
Section 61.56(a)(8) Waste containing hazardous, biological, pathogenic, or infectious material must be treated to reduce to the maximum extent practicable the potential hazard, from non-radiological materials	(No entry)
Section 61.56(b)(1) Waste must have structural stability. A structurally stable.... Section 61.56(b)(3) Void spaces within the waste and between the waste and its package must be reduced to the extent practicable.	Section 3.i.5.a ...for all types of containers, void spaces within the waste and between the waste and its packaging shall be reduced as much as practical. (At DOE Sites, site specific waste acceptance criteria, mandated by Section 3.e of the order, deal with stability requirements.)
Section 61.29 Following completion of closure...the licensee shall observe, monitor and carryout...maintenance...until the license is transferred.... ...the disposal site must be maintained...for 5 years. A shorter or longer time period for post closure observation and maintenance may be established and approved as part of the site closure plan, based onsite-specific conditions.	Section 3.j.1 Field organizations shall develop site specific comprehensive closure plans for...disposal sites. The plan shall address closure...within a 5 year period after each is filled... Section 3.j.9 Termination of monitoring and maintenance activity at closed facilities or site shall be based on an analysis of site performance at the end of the institutional control period.

TECHNICAL BASIS FOR CLASSIFICATION OF LOW-ACTIVITY WASTE FRACTION FROM HANFORD SITE TANKS

For the

Tank Waste Remediation System

September 1996

Originally prepared by:

Disposal Engineering
Westinghouse Hanford Company
Richland, Washington

Revised by:
Los Alamos Technical Associates, Inc. and
BNFL, INC.
Richland, Washington

Prepared for:
U.S. Department of Energy
Richland, Washington

9803170216 980306
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WHC-SD-WM-TI-699, Rev. 2

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EXECUTIVE SUMMARY

INTRODUCTION

The Hanford Site tank wastes were produced through the years 1944 to 1988 by reprocessing irradiated nuclear fuel (containing 98,100 Mg of uranium). The resulting aqueous wastes were accumulated in underground storage tanks. Over the years, these tank wastes were treated to reduce the volume, to minimize the volume of liquid in single-shell tanks (SSTs), and to remove a portion of the radionuclides.

In 1989, the U.S. Nuclear Regulatory Commission (NRC) evaluated a U.S. Department of Energy (DOE) proposal for pretreatment and disposal of Hanford Site double-shell tank (DST) reprocessed liquid wastes (Bernero 1989) and agreed that, if treated as described by DOE, a portion the wastes would be classified as "incidental" wastes. This classification would remove the waste from the high-level waste (HLW) disposal licensing authority of the NRC and would permit disposal of the wastes under DOE requirements in shallow land disposal facilities. This proposal was based on the preferred alternative in the *Environmental Impact Statement for the Disposal of Hanford Defense, Tank and Transuranic Wastes* (HDW-EIS) (DOE 1987). The proposal included removal of ^{137}Cs from Neutralized Current Acid Waste and Complexant Concentration (CC) waste supernatants and immobilizing the resultant liquid as a grout for permanent disposal in near-surface vaults. The proposal also included transuranic (TRU) element removal from Neutralized Cladding Removal Waste and Plutonium Finishing Plant wastes sludges. SSTs were not included to allow further study on "if" and "how" SST wastes would be disposed.

Since completion of the original agreement between the NRC and the DOE, several changes have occurred that impact the tank waste processing plan. The main changes include plans to retrieve SST waste and concerns about grout as a low-activity waste (LAW) form. These changes have occurred based on public and governmental agency review of the processing scheme for disposal of Hanford Site tank wastes. In addition, the DOE is pursuing privatization of the tank waste pretreatment and immobilization functions at the Hanford Site. Because of these changes, and additional information acquired from tank waste characterization, DOE believes it now has information sufficiently developed and documented to revisit the DST waste classification and to seek SST waste classification.

REPORT OBJECTIVE

The overall objective of this report is to provide a technical basis to support an NRC determination to classify the LAW (existing supernatants, plus wash solutions and dissolved salt cake) from the Hanford Site single-shell and double-shell tanks as "incidental" wastes after removal of additional radionuclides and immobilization. The proposed processing method, in addition to the previous radionuclide removal efforts, will remove the largest practical amount of total site radioactivity, attributable to HLW, for disposal in a deep geologic repository. The remainder of the waste would be considered "incidental" waste and could be disposed onsite.

This report assesses the degree of radionuclide removal from the tank liquid wastes against the following three guidelines (Bernero 1993):

1. The "waste has been processed (or will be further processed) to remove key radionuclides to the maximum extent technically and economically practical."

2. The "waste will be incorporated in a solid physical form at a concentration that does not exceed the applicable concentration limits for Class C low-level waste as set out in 10 CFR Part 61."

3. The solid, immobilized waste will be managed, pursuant to the *Atomic Energy Act of 1954*, so that safety requirements comparable to the performance objectives set out in 10 CFR Part 61 are satisfied.

Key radionuclides are considered to be: ^{137}Cs, ^{90}Sr, TRU, ^{99}Tc, ^{79}Se, ^{14}C, ^{129}I, ^{3}H, ^{126}Sn, and the uranium isotopes. These radionuclides are of interest because they represent 99.9 percent of the inventory, are specifically identified in 10 CFR Part 61, or are potential detractors to disposal system performance because they (or their daughter products) may exceed the dose limits for a short term intruder scenario (100 mrem/yr) or a long term drinking water scenario (4 mrem/yr). The key radionuclides listed are candidates for removal to the maximum extent technically and economically practical.

SUMMARY OF RESULTS

Of the estimated 422 MCi of key radionuclides that entered the Hanford Site tanks, 243 MCi have decayed during storage and approximately 87 MCi have been removed by previous processing, the bulk of which will eventually be disposed in a deep geologic repository. The remaining 92 MCi are addressed in this report.

Sludge washing and solids/liquid separation operations are intrinsic to all planned processing scenarios for separation of the high-activity sludges from the liquid phase. Separation of solids from supernatants by centrifuges and filters has been practiced in production operations at the Hanford Site since the 1940's. Solid/liquid separation by in-tank settle/decant has been practiced in tank farm operations over the last 50 years. Solids/liquid separation, which includes enhanced sludge washing, of the retrieved tank wastes will leave approximately 54 MCi in the solids slurry to be treated as HLW and approximately 38 MCi in the low-activity liquid phase before any additional radionuclide removal.

Cesium represents 91 percent of the curies (34 MCi soluble cesium) in the liquid fraction after solids/liquid separations have been performed. Removal of ^{137}Cs from the liquid fraction for return to the HLW fraction is technically and economically practical.

For this analysis, the ^{137}Cs removal is based on the use of an "in-plant" cesium ion exchange technology considered to have limited technical risk but relatively high cost. Assuming a single-cycle ion exchange system with a design basis of 99 percent theoretical removal and an overall operational efficiency of 97 percent, an estimated 33 MCi ^{137}Cs is technically

practical to remove from the liquid fraction. However, an evaluation of the cost to remove cesium from all the retrieved wastes shows that for dilute feeds (cesium concentration <0.05 Ci/L) the cost of further curie removal increases dramatically making further removal not economically practical. Processing of all feeds >0.05 Ci ^{137}Cs per liter would leave 5 MCi remaining in the LAW (i.e., 29 MCi ^{137}Cs separated).

Removal of TRU and ^{90}Sr from all the tank waste is not economically practical. However, selective removal of TRU from three tanks will be necessary to meet the Class C TRU limit. Laboratory testing on actual tank wastes indicate that the liquid fraction can be treated to produce < 100 nCi/g LAW glass by hydroxide adjustments. There are also indications that some of the soluble ^{90}Sr will co-precipitate during this hydroxide adjustment process. Additional ^{90}Sr removal is not economically practical. The estimated concentration of ^{90}Sr in the LAW feed is below the Class C limit for ^{90}Sr.

Technology exists for removal of ^{99}Tc, but is considered not to be technically practical. The estimated maximum concentration of ^{99}Tc in the LAW feed is within the Class A limit for ^{99}Tc. However, removal of ^{99}Tc may be required to meet disposal system performance objectives. A performance assessment will identify areas of concern for long-term release of radionuclides to the environment from the disposal system. Disposal design features will be evaluated that include waste form corrosion rate, waste form dimensions, engineered barriers, modification of water chemistry, chemical retardants, and moisture diverters. Upon completion of the disposal system evaluation, mitigating measures, such as ^{99}Tc removal and/or disposal system barriers, will be finalized.

No technologies have been adequately demonstrated for removal of ^{3}H, ^{14}C, ^{79}Se, or ^{129}I that can be considered technically practical. These are low concentration isotopes and some will most likely be removed from the LAW in the offgas system as a result of the vitrification process. Some ^{79}Se removal may be required to meet disposal system performance objectives.

CONCLUSION

The proposed new determination would include processing the tank wastes to accomplish the following:

1. Radionuclide removal to the maximum extent technically and economically practical will leave no more than 5 MCi ^{137}Cs and 3.4 MCi ^{90}Sr in the LAW.

2. Remove TRU as required (i.e., removal from 3 CC tanks) to ensure all solidified LAW is < 100 nCi TRU/g.

3. Meet all disposal requirements including those defined by the performance assessment.

It is concluded that cesium removal from liquid supernatants and TRU removal from CC tanks before LAW immobilization[1] represents radionuclide removal to the maximum extent practical. Therefore, these liquids could be considered "incidental" waste provided the immobilized LAW qualify for disposal in a shallow land disposal facility under DOE waste management requirements comparable to 10 CFR Part 61 requirements.

[1]This report assumes that the LAW form for disposal is a glass waste form. Therefore, the concentrations of the radionuclides are concentrations using a vitrification process.

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LIST OF TERMS AND DEFINITIONS

Terms

ANL	Argonne National Laboratory
ANPR	Annual Notice of Proposed Rulemaking
CAW	Current acid waste
CC	Complexant concentrate
CFR	Code of Federal Regulation
CST	Crystalline silicotitanate
DF	Decontamination factor
DOE	U.S. Department of Energy
DSS	Double-shell slurry
DSSF	Double-shell slurry feed
DST	Double-shell tank
Ecology	Washington State Department of Ecology
EIS	Environmental Impact Statement
ETF	Effluent Treatment Facility
FY	fiscal year
HAPO	Hanford Atomic Products Operations
HDW-EIS	*Final Environmental Impact Statement for the Disposal of Hanford Defense, High-Level Transuranic and Tank Wastes*
HLW	High-level waste
LAW	Low-activity waste
LLW	Low-level waste
NCAW	Neutralized current acid waste
NRC	U.S. Nuclear Regulatory Commission
PA	Performance Assessment
PFP	Plutonium Finishing Plant
PUREX	Plutonium-uranium extraction
REDOX	Reduction Oxidation
SALDS	State Approved Land Disposal Site
SOLEX	Solvent extraction
SREX	Strontium extraction
SRS	Savannah River Site
SST	Single-shell tank
TBP	Tributyl phosphate
TRU	Transuranic
TRUEX	Transuranic extraction
TWRS	Tank Waste Remediation System
WESF	Waste Encapsulation and Storage Facility
WHC	Westinghouse Hanford Company
WVDP	West Valley Demonstration Project

Definitions

High-level waste: For the purposes of this document, HLW is defined as those aqueous wastes resulting from the operation of the first-cycle solvent extraction system, or equivalent, and the concentrated wastes from subsequent extraction cycles, or equivalent, in a facility for reprocessing irradiated reactor fuels (10 CFR Part 50, Appendix F).

Incidental waste: The term incidental waste originated when the 10 CFR Part 50, Appendix F definition for HLW was promulgated and the Atomic Energy Commission specifically noted that the term HLW did not include "incidental" waste resulting from reprocessing plant operations, such as ion exchange beds, sludges, and contaminated laboratory items, such as clothing, tools, and equipment. Under the same reasoning, the NRC has indicated incidental wastes generated in further treatment of HLW (e.g., salt residues or miscellaneous trash from waste glass processing) would be outside the Appendix F definition.

For application to the Hanford Site wastes, incidental wastes include the miscellaneous wastes resulting from further processing of HLW to enhance the product (e.g., volume reduction) or to remove nonradioactive materials previously added to the HLW (e.g. neutralization of acidic HLW).

Liquid fraction: The low-activity liquid fraction of the tank waste containing a small fraction of water-insoluble solids.

Low-level waste: As specified in the Low-level Radioactive Waste Policy Amendments Act of 1985 (Public Law 99-240), radioactive waste not classified as high-level waste, spent nuclear fuel, or by-product material specified as uranium or thorium tailings and waste.

Low-activity waste: Low-activity waste (LAW) at the Hanford Site is produced by treating the tank wastes. LAW is produced by treating the tank wastes and are "low-level" tank wastes that have not yet received the NRC concurrence as incidental.

Sludge washing: Contacting tank waste sludges with dilute sodium hydroxide solutions to dilute and dissolve aqueous soluble components. Enhanced sludge washing contacts sludges with 3 molar sodium hydroxide solutions to dissolve selected components (aluminum, chromium, phosphorous, and sodium).

Supernatant: The liquid layer above the solids in the waste storage tanks. The liquid supernatant includes drainable interstitial liquids.

Transuranic waste: As defined and used by the U.S. Department of Energy (DOE Order 5820.2A), radioactive waste that, at the time of assay, contains more than 100 nCi/g of alpha-emitting isotopes with atomic numbers greater than 92 and half-lives greater than 20 years.

Other notes:

Units Reported: Throughout the document, all characteristics (i.e., volume, mass, etc.) are reported in metric units except for the radiological units of Sievert (Sv) and Becquerel (Bq). In these two cases, the English units will be used: mrem and curie (Ci), respectively. In some instances, both metric and english units are shown.

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TECHNICAL BASIS FOR CLASSIFICATION OF LOW-ACTIVITY WASTE FRACTION FROM HANFORD SITE TANKS

1.0 INTRODUCTION

National defense activities have generated radioactive waste since 1944 on the Hanford Site in Washington State. Liquid radioactive and chemical wastes from the nuclear material production and research activities were transferred to underground, reinforced-concrete, steel-lined tanks [commonly referred to as single-shell tanks (SSTs) and double-shell tanks (DSTs)] for storage. The U.S. Department of Energy (DOE) and its predecessor organizations (the Manhattan Engineering District, Atomic Energy Commission and Energy Research and Development Administration) discharged waste from reprocessing into SSTs from 1944 to 1971 and into DSTs from 1971 to 1988. Active use of the SSTs ceased in November 1980. Since 1980, only stabilization and isolation activities have occurred in the SSTs. Plans are to retrieve waste from both single- and double-shell tanks, pretreat the waste as necessary to separate high-level waste (HLW) and low-activity waste (LAW) fractions and immobilize both fractions by vitrification.

This document provides an analysis to support a determination of the radioactive classification of the LAW fraction produced by treating Hanford Site tank wastes. To allow for disposal of the LAW fraction of tank waste near surface, the DOE must receive a determination from the U.S. Nuclear Regulatory Commission (NRC) that the waste is not HLW and, as such, is not subject to NRC licensing.

1.1 OBJECTIVE

The objective of this report is to provide a technical basis to support a NRC determination to classify the LAW fraction of Hanford Site tank waste as "incidental" waste.

1.2 SCOPE

Based on the current tank waste processing scheme, this document evaluates removal of key radionuclides from the liquid phase of tank wastes to produce a LAW fraction for onsite disposal. Consistent with the revised processing plan for Hanford Site tank wastes (Ecology et al. 1994), the document discusses: (1) the current NRC classification criteria and requirements, (2) the tank waste inventory and radionuclides of interest (i.e., "key" radionuclides), (3) technology options for treating liquids to produce a LAW fraction suitable for onsite disposal and, (4) the technology status and costs for separations processes.

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2.0 METHODOLOGY

This section provides a description of the process for designating a waste as "incidental" and describes the application of the NRC process to the Hanford Site tank waste. Although the NRC does not have licensing authority for the disposal of DOE's low-level waste (LLW), the NRC does have authority for licensing the disposal of HLW for both commercial and defense waste. LAWs are produced by treating the tank wastes and are "low-level" tank wastes that have not yet received the NRC concurrence as incidental. To allow for disposal of the LAW fraction of tank waste near surface, the DOE must receive a determination from the NRC that the waste is not HLW and, as such, is not subject to NRC licensing.

2.1 WASTE CLASSIFICATION FOR THE HANFORD SITE DOUBLE-SHELL TANK WASTE

The NRC and the DOE have established a process for determining what defense waste constitutes "incidental" waste. They have applied this process to the Hanford Site DST wastes, Savannah River (South Carolina) wastes and West Valley (New York) tank wastes.

For the Hanford Site tank wastes, the NRC previously concluded that the reprocessing wastes to be disposed in the grout facility (e.g., DST supernatants) would be classified as "incidental" wastes. This classification was based on the processing scheme defined in the *Final Environmental Impact Statement for the Disposal of Hanford Defense, High-Level, Transuranic and Tank Wastes, Hanford Site, Richland, Washington; Record of Decision* (HDW-EIS) (DOE 1988). Subsequent to this classification, the *Hanford Federal Facility Agreement and Consent Order* (Tri-Party Agreement) (Ecology et al. 1994) was enacted and the processing scheme for treatment of the tank wastes was modified. The modified processing scheme includes retrieval and processing of single-shell tanks, not to use existing facilities (i.e., B Plant) for tank waste pretreatment, and to dispose of the LAW fraction as glass instead of grout.

The NRC determination for Hanford Site DST wastes was provided in Bernero 1989 and confirmed in 58 FR 12342 (documentation contained in Appendix A). A review of the NRC determination and the subsequent modification of the tank processing scheme is provided in the following section.

2.1.1 U.S. Nuclear Regulatory Commission Determination for Hanford Site Double-Shell Tank Waste

The NRC evaluated and accepted the DOE proposal for pretreatment and disposal of Hanford Site DST wastes (Bernero 1989). The DOE based this proposal on the preferred option from the *Environmental Impact Statement for the Disposal of Hanford Defense, Tank, and Transuranic Wastes* (HDW-EIS) (DOE 1987). The DOE proposal (Rizzo 1989) is included in Appendix A.

The states of Oregon, Washington, and the Yakama Indian Nation petitioned the NRC to exercise its rulemaking authority and adopt a regulation concerning classification of Hanford Site high-level radioactive wastes currently stored in retrievable surface, storage facilities (e.g., DSTs). In addition to the rulemaking request, the petitioners requested application of the radionuclide separation criteria on a tank-by-tank basis. In response to the petition, the NRC reviewed their original finding and obtained comments from the public. In 58 FR 12342, the NRC found that DOE's plans for the handling of DST wastes were consistent with their principles of waste decontamination and protection of the public, and denied the rulemaking request.

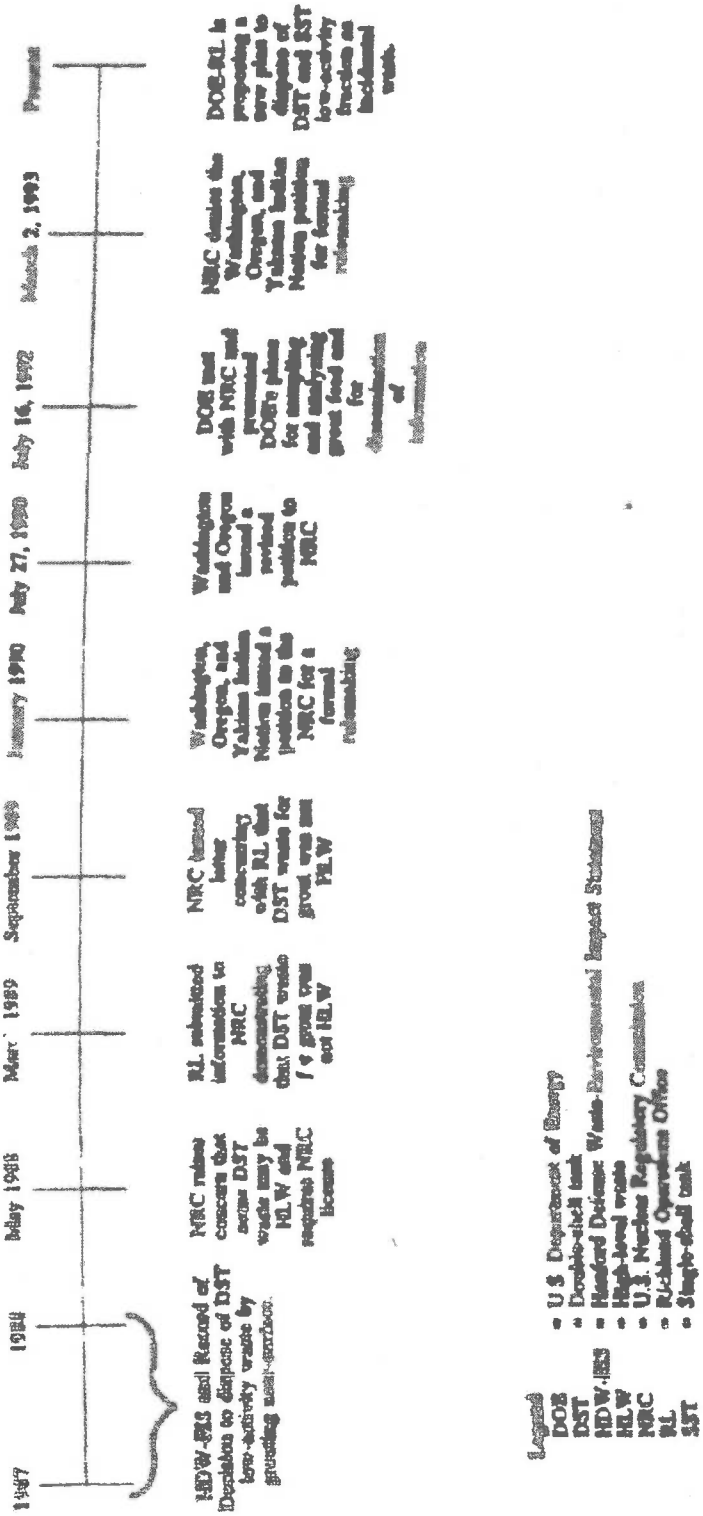
Figure 2-1 shows a historical timeline on the classification of the LAW fraction from the Hanford Site DST waste. This timeline begins with the completion of the *Final Environmental Impact Statement–Disposal of Hanford Defense High Level, Transuranic, and Tank Wastes* (HDW-EIS) and ends with the NRC confirmation in 1993 (58 FR 12342). Appendix A contains the correspondence associated with the timeline as well as a synopsis of the correspondence.

2.1.2 Modification of the Tank Waste Processing Scheme

Since completion of the original agreement between the NRC and the DOE, several changes have occurred (Wodrich 1994) which impact the tank waste processing scheme. These changes occurred based on a review of the processing scheme with the public and governmental agencies involved in the disposal of tank waste. The primary changes that impact the LAW determination include: (a) planned retrieval of all SST waste; (b) an existing plant will not be used as a pretreatment facility; and (c) a change from grout to glass as a LLW form.

(a) Since the HDW-EIS (DOE 1987) was issued, the knowledge and the conditions surrounding waste disposal have changed considerably, reversing previous thinking that favored in situ disposal for the SSTs. The emerging tank waste safety issues have raised concerns about long-term stability. Also, the *Resource Conservation and Recovery Act of 1976* and the *Nuclear Waste Policy Act of 1982* requirements for treatment and geologic disposal of HLW run counter to the in situ disposal approach for SSTs.

Figure 2-1. Timeline of Classification of Low-Activity Waste Fraction in Hanford Site Tanks.



As a result, it is now planned to retrieve and process all DST and SST waste for disposal. Currently, the SST waste is mostly salt cake or sludge. However, the waste in the SSTs must now be redissolved or made into a liquid slurry, retrieved, transferred, and fed into the pretreatment process system. This change increases by four-fold the overall amount of waste requiring processing and the amount of waste for processing to remove cesium increases from about 20,000 m^3 to over 600,000 m^3 if all tank waste is processed.

(b) Another primary change was based on a technical evaluation that found the B Plant to be unacceptable for use as a pretreatment facility. It was determined B Plant did not comply with current environmental and safety regulatory criteria (Grygiel et al. 1991). Upgrades to reach compliance are not technically or economically practical. The decision not to use B Plant for pretreatment increases the time for implementing radionuclide separations since new facility construction is required.

(c) Grout, as planned for near-surface disposal, was included in the HDW-EIS Record of Decision (DOE 1988). However, stakeholders were concerned about the performance of grout for immobilizing LAW over a long period of time (leachability) and life-cycle cost estimates for other LAW forms being no greater than grout. Other factors such as radionuclide content, lack of retrievability, and the large volume of grout, added to stakeholder concerns about continuing with grout.

To accommodate these changes, the DOE initiated a program rebaselining activity to re-evaluate and plan a revised approach to disposal of the Hanford Site tank waste. The elimination of B Plant as a potential facility for development of pretreatment separations technology, as well as for future pretreatment operations, made the 1989 Tri-Party Agreement milestones difficult to accomplish on schedule.

In December 1991, the Secretary of Energy directed that an integrated Tank Waste Remediation System (TWRS) Program be established to plan and implement the disposal of the tank waste at the Hanford Site. A rebaselined TWRS Program mission, based on a proposed new technical strategy, was provided by DOE in March 1993 to the other two parties of the Tri-Party Agreement in the form of a request to modify the agreement. The tri-parties modified the agreement and signed the fourth amendment to the Tri-Party Agreement (Ecology et al. 1994).

The current fiscal year (FY) 1996 planning calls for a two-phase approach to waste processing (Bader 1995). In 1995, the waste treatment and immobilization acquisition strategy was changed to privatize these functions. This strategy would utilize private contractor(s) to design, build, operate and finance the facilities with DOE paying a fixed price for immobilized waste products delivered to DOE's specification. The plan calls for a proof-of-concept/commercial demonstration-scale effort (referred to as Phase I) to provide confidence in technology, funding methodology, and regulatory strategies to allow financing of a privatization contractor in the next phase. Following Phase I, the plan calls for a full-

scale production phase (referred to as Phase II). In Phase I and Phase II, DOE will be purchasing services from a contractor-owned, contractor-operated facility under a fixed-price type of contract. It should be noted that the contractor(s) for Phase I and II of the TWRS privatization strategy may select radionuclide removal technologies and/or a LAW form different than presented in this document.

2.2 NUCLEAR REGULATORY COMMISSION WASTE CLASSIFICATION

A clear distinction between HLW and LLW does not exist. High-level waste has a source based definition while LLW is generally defined by what it is not (e.g., HLW). The NRC classification of Hanford Site tank wastes is defined by the *Low-Level Radioactive Waste Policy Act* and the definitions for HLW and "incidental" waste in 10 CFR Part 50, Appendix F.

2.2.1 Definition of Low-Level Waste

The current definition of LLW for both the NRC (10 CFR Part 61) and DOE (DOE Order 5820.2A) comes from the *Low-Level Radioactive Waste Policy Act* The definition designates LLW as radioactive waste that is not classified as HLW, transuranic (TRU) waste, or nuclear fuel or byproduct material as defined in 11e.(2) of the *Atomic Energy Act of 1954* (uranium and thorium tailings and waste). Thus, to determine whether a waste is LLW it must not meet the characteristics of the waste types listed.

The NRC places concentration restrictions on the disposal of waste near surface. Concentration limits for disposal of commercial LLW (which must be in a solid form) are specified in 10 CFR Part 61.55, Waste Classification Requirements for Land Disposal Facilities. The NRC issued an Advance Notice of Proposed Rulemaking (ANPR) for a concentration based Definition of "High-Level Radioactive Waste," 52 FR 5992. Under this ANPR, a waste having concentrations less than either of those found in Table 1 or Table 2 from 10 CFR Part 61 would constitute LLW. Public and agency comment on this ANPR led to recision of the proposal as documented in 53 FR 17709.

The release of radionuclides to the environment is addressed in 10 CFR Part 61.41, which requires that the disposal action does not exceed a 250 μSv per year (25 mrem/yr) dose from all pathways.

The DOE does not place quantitative concentration restrictions on the disposal of LLW, but requires completion of a performance assessment of the waste disposal activity per DOE Order 5820.2A. The performance assessment must show that the disposal action provides adequate protection to the groundwater per 40 CFR Part 141. Complying with this requirement mandates that the disposal action does not exceed a 40 μSv per year (4 mrem/yr) dose from the groundwater pathway.

2.2.2 Definition of High-Level Waste

Appendix F of 10 CFR Part 50 codified what the Atomic Energy Commission deemed HLW in 1970. For defense waste, the NRC utilizes this definition which was in place when they were given jurisdiction for its long term storage in 1974. This definition consists of a source based definition for HLW. HLW is defined as "those aqueous wastes resulting from the operation of the first-cycle solvent extraction system or equivalent, and the concentrated waste from subsequent extraction cycles, or equivalent, in a facility for reprocessing irradiated reactor fuels."

Another definition of HLW is contained in the 1987 amendment to the *Nuclear Waste Policy Act of 1982* (NWPA). The NWPA defines HLW in the following two parts:

1. "The highly radioactive material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentration; and

2. Other highly radioactive material that the Commission, consistent with existing law, determines by rule requires permanent isolation."

In context of the NWPA, "requires permanent isolation" means disposal in geologic repository or alternative system with equivalent waste-isolation capabilities.

2.2.3 U.S. Nuclear Regulatory Commission Guidelines for Classification of Hanford Site Wastes

The NRC position for Hanford Site wastes was given by Bernero (1989). "At Hanford, the question of waste classification (and NRC licensing authority) has been complicated by the mixing of waste from various sources over the past 45 years. This mixing has changed the original characteristics of the wastes and has resulted, in some cases, in the mixing of HLW and low-level waste (LLW). Consequently, it is now difficult to directly differentiate between HLW and LLW, using the source-based definition of 10 CFR Part 50, Appendix F."

The rulemaking record for Appendix F specifically recognizes a number of "incidental," non-HLW waste streams associated with reprocessing plant operations. These include cladding hulls, ion exchange media, sludges and miscellaneous trash generated during reprocessing operations. Not mentioned, however, are wastes resulting from further processing of HLW (e.g., volume reduction) or removing non-radioactive materials that were added to the HLW for improved processing and/or storage (e.g., the addition of alkaline material to neutralize acidic HLW). At West Valley and the Savannah River Plant, NRC has agreed that such wastes are not HLW.

The current NRC classification for Hanford Site DST wastes was confirmed in 58 FR 12342. In this determination, the NRC concluded that DOE's proposed processing scheme would remove the largest practical amount of the total site activity attributable to HLW, for disposal in a deep geologic repository, and the remainder of the waste would be considered "incidental" waste and could be disposed of onsite.

The Commissions's conclusion that the reprocessed DST wastes would be "incidental" wastes was based on DOE's assurances that they have met the following guidelines (Bernero 1993):

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

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

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

Figure 2-2 depicts these steps as actions and decisions in the form of a logic diagram.

2.3 INTERPRETATION OF U.S. NUCLEAR REGULATORY COMMISSION GUIDELINES

The first NRC guideline outlined in Section 2.2.3 uses the criteria "technically and economically practical." Technically and economically practical, as applied to this study, is defined below. NRC guidelines for meeting the concentration limits for Class C LLW and performance objective requirements as set out in 10 CFR Part 61 are well-defined and not as subject to interpretation as the criteria for "technically and economically practical." The application of these guidelines to evaluate specific radionuclide separation processes is provided in Chapter 3.0.

2.3.1 Definition of Technically Practical and Economically Practical

Technically practical radionuclide removal technology is defined herein as a technology or process option having plant scale experience or a high probability for successful operation. High probability of success is ensured through laboratory testing on actual tank wastes, applications in like or similar missions, and an engineering judgement of the difficulty to scale a process option for full-size application (e.g., complexity of operation).

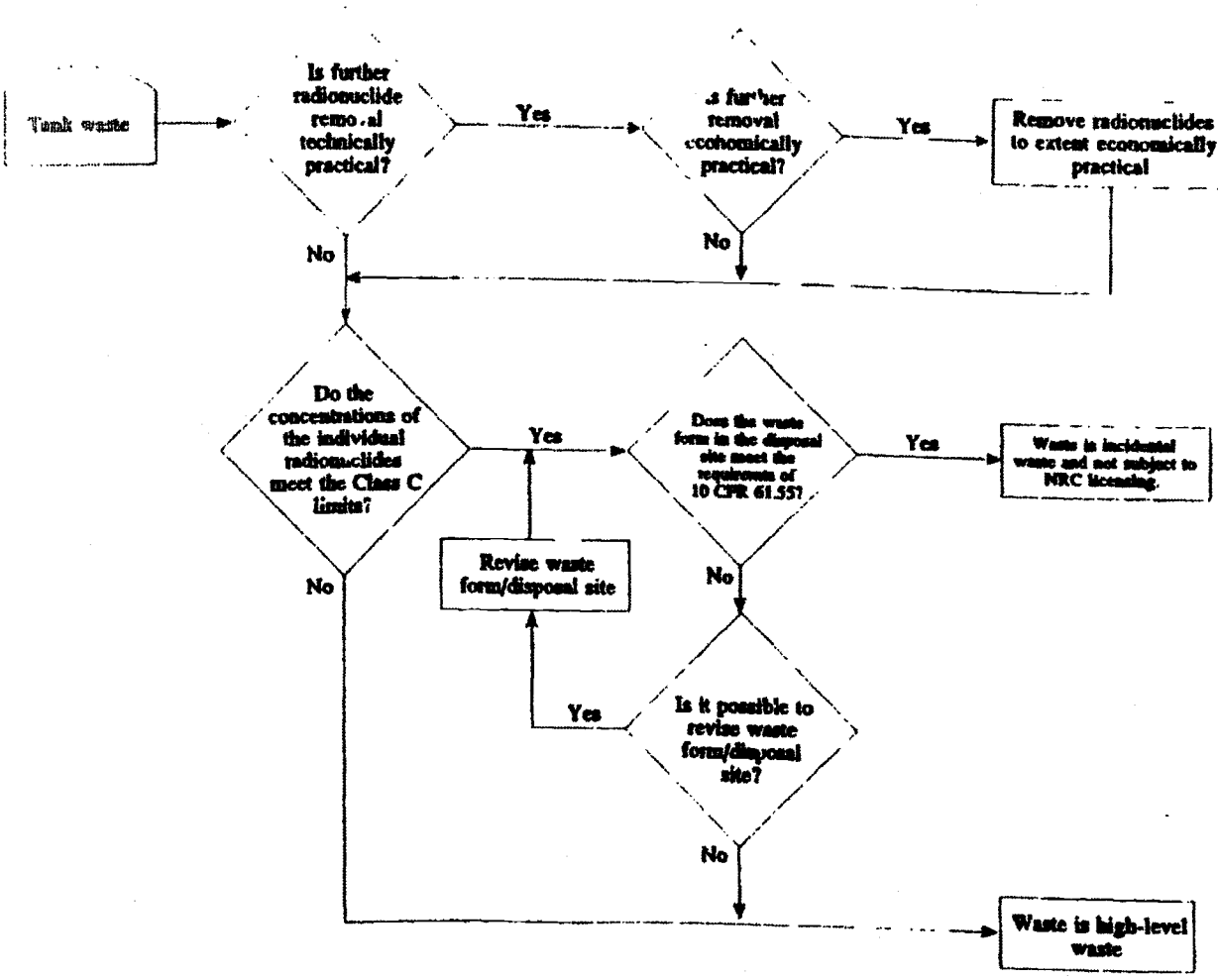


Figure 2-2. Logic Diagram for Determining Classification of Tank Waste.

Economic practicality is determined by the total life-cycle cost and the cost per curie removed. The economically practical limit is selected for this evaluation as the point where additional removal costs increase significantly. An economic assessment is not provided if a technology option is considered to be "not technically practical."

2.3.2 10 CFR Part 61 Class C Concentration Limits

The second NRC guideline for classification of the LAW fraction as "incidental" waste is to incorporate the waste in a solid physical form at a concentration that does not exceed the applicable concentration limits for Class C LLW as set out in 10 CFR Part 61.55.

2.3.3 10 CFR Part 61 Disposal Performance Objectives

The third NRC guideline for classification of the LAW fraction as "incidental" waste is to manage the waste so that safety requirements for disposal performance are comparable to the performance objectives of 10 CFR Part 61 (less than 25 mrem/yr by all pathways). Defense wastes are managed and disposed under guidance from DOE 5820.2A, which also requires less than 25 mrem/yr by all pathways and compliance with 40 CFR Part 141 (less than 4 mrem/yr exposure from groundwater). The DOE order also requires that a Performance Assessment (PA) be prepared for the disposal system. A PA for LAW disposal has not yet been prepared. Appendix D is a comparison of the disposal performance requirements in 10 CFR part 61 and DOE 5820.2A.

A study of disposal system design feature impacts on LAW disposal system performance (Mann et al. 1995) indicates that some ^{99}Tc and ^{79}Se removal and/or disposal design features may be required to meet the NRC guideline and DOE requirements. Disposal design features addressed in the study include waste form corrosion rate, waste form dimensions, engineered barriers, modification of water chemistry, chemical retardants, and moisture diverters. When the PA is complete it will identify areas of concern for long-term release of radionuclides to the environment from the disposal system. At that time, mitigating measures, such as additional radionuclide removal and/or disposal system barriers, will be evaluated and incorporated into the treatment and disposal system as needed to meet PA requirements.

2.3.4 Radionuclides of Interest

The radionuclides of interest in meeting the NRC guidelines for classification of the LAW fraction as "incidental" waste are shown in Table 2-1. These radionuclides are of interest because they represent 99.9 percent of the inventory or are specifically identified in 10 CFR Part 61 (Tables 1 and 2). The radionuclides of interest for the disposal system performance are also shown in Table 2-1. Table 2-1 includes the top five radionuclides for intruder consequences and the top 5 radionuclides based on dose consequences from a drinking water scenario which includes the effect of soil retardation (Schmittroth et al. 1995). An interim performance assessment is in progress and the radionuclides listed are being evaluated to determine if consequences exceed dose for a short term intruder scenario (100 mrem/yr) and/or a long term drinking water scenario (4 mrem/yr). To ensure potentially significant isotopes continue to be monitored, the Interim Performance Assessment also includes additional radionuclides other than those listed in Table 2-1. These additional radionuclides are considered unlikely to be significant for the PA and therefore are not key radionuclides for this study.

Table 2-1. Radionuclides of Interest in meeting the U.S. Nuclear Regulatory Commission "Incidental" Waste Guidelines.

Radionuclide*	Exceed 10 CFR Part 61 Class C limit	Potentially may exceed disposal performance objectives	
		Intruder dose consequence	Groundwater dose consequence
^{137}Cs	2 tanks	X	
^{90}Sr		X	
Transuranics	3 tanks	X	
^{99}Tc			X
^{3}H			
^{14}C			
^{129}I			
^{126}Sn		X	
^{79}Se			X
Uranium Isotopes (^{233}U, ^{235}U, ^{238}U)			X

*^{63}Ni, ^{60}Co, and ^{94}Nb are listed in 10 CFR Part 61 but are not included as key radionuclides. Due to the small inventory of these radionuclides, no significant contribution is made that would affect the determination of the waste classification.

3.0 SOURCE OF TANK WASTE

The Hanford Site tank wastes were produced through the years 1944 to 1988 by reprocessing irradiated nuclear fuel (containing 98,100 Mg of uranium). The aqueous wastes produced were accumulated in 177 underground storage tanks. During this period, these wastes were treated to reduce the stored volume and to remove a portion of the radionuclides. Figure 3-1 depicts the estimated total curie inventory that has been added to the tanks and the number of curies remaining in the tanks (DOE 1994a).

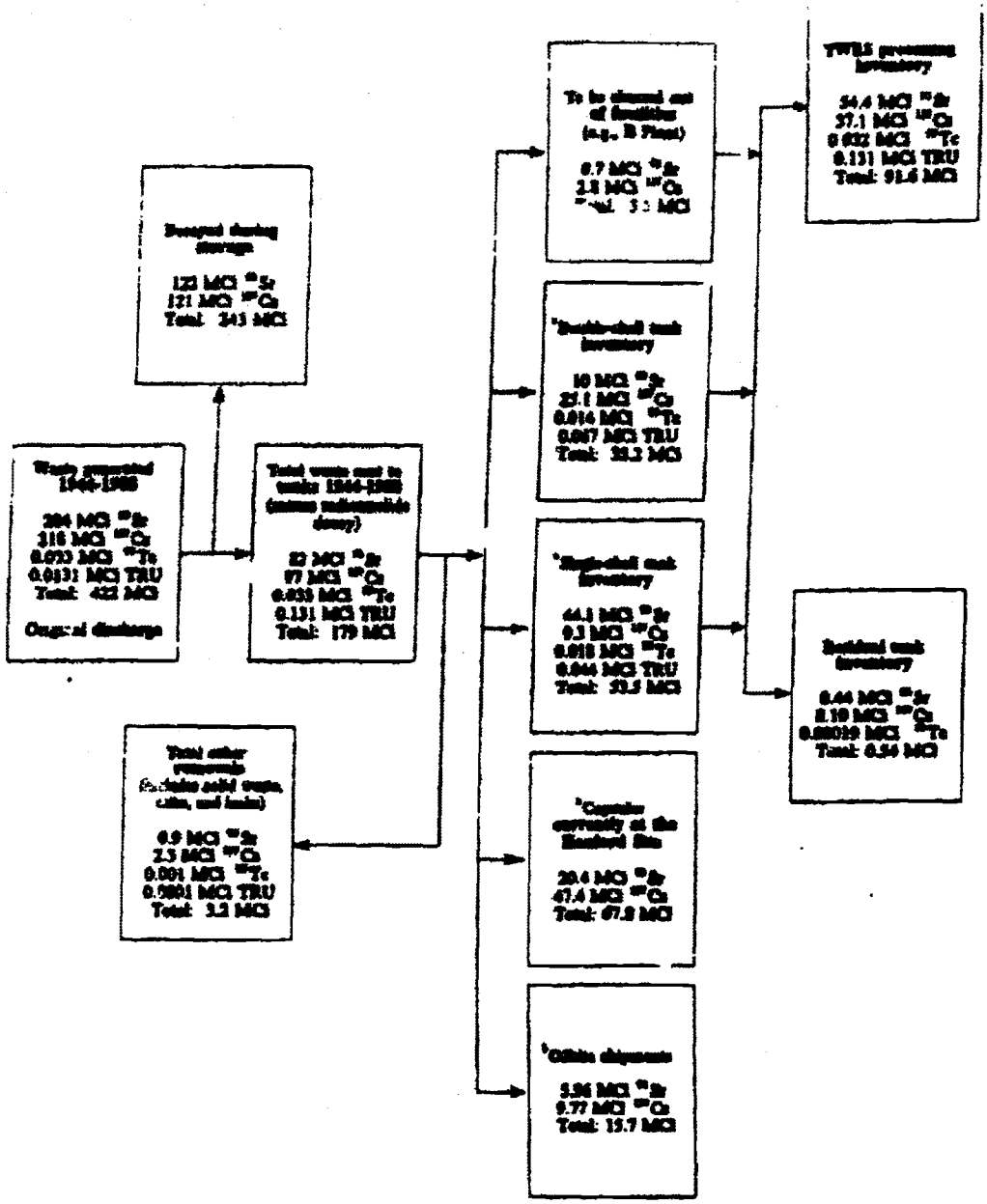
The existing liquid fraction, dissolved salt cake and liquids from treatment of sludges/solids are the candidates for disposal as LAW. Salt cake, generated as a result of supernatant concentration, contains almost none of the soluble radionuclides. Residual interstitial liquor in the salt contains a relatively high cesium concentration but the total volume is small. Therefore, the existing liquid waste fraction contains the overwhelming inventory of the soluble radionuclide inventory.

The solid fraction (sludge) and those radioisotopes separated from the liquid fraction will be vitrified as HLW and disposed in a geologic repository. The insoluble solids (sludges) contain the bulk of the ^{90}Sr, TRU, and a small amount of ^{137}Cs. The predominant radionuclide in the liquid is ^{137}Cs with a small amount of ^{90}Sr and lesser fractions of TRU, ^{99}Tc, ^{3}H, ^{14}C, ^{79}Se, and ^{129}I. The cesium and strontium previously removed from the waste are presently stored in capsules and are not currently considered to be waste. Capsules, should they be considered waste, will be disposed as HLW in the geologic repository.

3.1 TANK WASTE INVENTORY FOR PRETREATMENT

Table 3-1 presents the total inventory of radionuclides that will be processed by pretreatment and are the values used to assess radionuclide removal requirements in Section 4.0. The tank waste inventory represents the amount of waste and contained radionuclides that will be processed by pretreatment and LAW and HLW vitrification. The TWRS processing inventory consists of the current tank inventory (Orme 1995) and future additions less the tank heel following waste retrieval. Appendix C provides additional information and references for the source of the tank waste inventory values.

Figure 3-1. Estimated Hanford Site Tank Waste Radionuclide Inventory.[c,d]



[a]Curie values are based on the Integrated Data Base Report-1994, Rev. 11, Table 2.11 decayed to December 31, 1999.

[b]The "offsite shipments" inventory is not expected to return to the Hanford Site for treatment.

[c]Decay products are not listed. Some radionuclides, such as ^{137}Cs and ^{90}Sr, have daughters with relatively short half-lives and are present in concentrations associated with the normal decay chain of the radionuclide.

[d]Inventories of other key radionuclides (i.e., ^{3}H, ^{129}I, ^{14}C, ^{79}Se, uranium isotopes, and ^{126}Sn) are not shown on the material balance. These radionuclides have small inventories that do not significantly affect the total curies in the material balance.

Table 3-1. Tank Waste Remediation System Processing Inventory (December 31, 1999, Decay Date).

Radionuclide	Tank Waste Remediation System Processing[a] Inventory, MCi		
	Soluble	Insoluble	Total
^{137}Cs	34.1	3.01	37.1
^{90}Sr	1.89	52.5	54.4
Transuranics	0.00961	0.121	0.131
^{99}Tc	0.0228	0.0093	0.032
^{79}Se	0.00103	-	0.00103
^{14}C	0.0053	-	0.0053
^{129}I	0.000051	-	0.000051
^{3}H	0.01	-	0.01
^{126}Sn	-	0.0016	0.0016
Uranium	0.00006	0.00094	0.001
Total	36.0	55.6	91.6[b]

[a]The inventories of ^{3}H, ^{14}C, and ^{129}I are given in Colby (1994). The inventory for ^{79}Se and uranium is given in Mann et al. (1995). The inventory for ^{126}Sn is given in Schmittroth et al. (1995). The primary source of tank waste inventories are given in Orme (1995) for ^{137}Cs, ^{90}Sr, and TRU because additional detail is given for fractions of soluble and insoluble radionuclides. The values for ^{137}Cs, ^{90}Sr, and TRU are consistent with the Integrated Data Base Report-1994. See Appendix C for source of TWRS processing inventory.

[b]Round-off error can result in $\pm$ 0.2 MCi.

3.2 INVENTORY UNCERTAINTIES

The inventories provided for evaluation of additional radionuclide removal are subject to uncertainties. A general discussion of known or anticipated limitations of the inventory values is given below. Resolution of inventory uncertainties is currently being addressed in Kupfer et al. 1995, draft only.

Cesium and Strontium Inventory: The reported inventories for ^{137}Cs and ^{90}Sr are expected to have small uncertainties (less than 10 percent).

Transuranics (includes ^{239}Pu, ^{240}Pu, ^{241}Am and ^{237}Np): The inventory uncertainty for the transuranics is primarily associated with the quantities in the insoluble fraction. This uncertainty does not affect an analysis of removal from the soluble fraction. The tank waste processing inventory of transuranics used for this analysis is consistent with the *Integrated Data Base Report*, Rev. 11.

Technetium Inventory: The ^{99}Tc inventories are based on the assumption of no removal of ^{99}Tc by previous processing. Previous technetium removals include cribbing as supernatants from the tanks, cribbing of process wastes during B Plant strontium and cesium recovery campaigns, ^{99}Tc recovery demonstration and shipment offsite, and removal from the Hanford Site as a contaminant in shipments of uranium oxide product. These previous removals may reduce the ^{99}Tc tank inventory by 25 to 50 percent (Colby and Petersen 1995). Analysis of the ^{99}Tc inventory is ongoing.

Selenium Inventory: The ^{79}Se inventories assume no removal of ^{79}Se by previous processing. Previous selenium removals include cribbing as supernatants from the tanks, and cribbing of process wastes during B Plant strontium and cesium recovery campaigns. These previous removals may reduce the ^{79}Se tank inventory by up to a factor of 2. Analysis of previous ^{79}Se removals is in progress.

Carbon Inventory: Because of the poorly known chemistry of ^{14}C in the fuel reprocessing operations that generated the Hanford Site tank wastes, the assumed inventory is conservative and the actual inventory may be a factor of 10 lower.

Iodine Inventory: Because of the poorly known removal of ^{129}I in the fuel reprocessing operations that generated the Hanford Site tank wastes, the assumed inventory is an upper bounding inventory and the actual inventory may be a factor of 2 to 10 lower.

Tritium Inventory: Tritium (^{3}H) contained in the tank wastes is estimated to be 10,000 Ci (Colby 1994). Tritium will be discharged, to a state approved disposal site, from the pretreatment and waste vitrification facilities in the process condensates as tritiated water. Analysis of the ^{3}H inventory is ongoing.

Tin Inventory: Some ^{126}Sn is expected to be solubilized in the alkaline solutions, but inventory values have not yet been specified. No significant quantity of ^{126}Sn is expected in the LAW that would affect the waste classification. Therefore, ^{126}Sn is not considered for additional radionuclide removal. For performance assessment studies, some tin to the LAW fraction is assumed to ensure continued consideration of tin for intruder dose consequences.

Uranium Inventory: The reactor discharges of the major uranium isotope, ^{238}U, are well established using the ORIGEN2 model. However, the production estimates of higher actinides, including other uranium and plutonium isotopes, is more difficult to calculate. Further analysis is needed to refine the values for ^{234}U, ^{236}U, and ^{241}Am.

Other–Sodium Inventory: The impact of a potential reduction in the tank sodium inventory was not quantitatively determined in this study. Qualitatively, the costs for cesium ion exchange will not change significantly with a reduction in the sodium inventory. Agnew (1995) indicates that the total sodium inventory in the tank wastes may be approximately 60 percent of the current reported values. This would decrease the predicted volume of the immobilized LAW form since sodium is the major constituent in the LAW, but will not affect Class C concentration limits.

4.0 RADIONUCLIDE SEPARATIONS TECHNOLOGIES

The TWRS plans to further pretreat the tank waste by separating the tank wastes into a HLW fraction and a LAW fraction. Pretreatment has the goal of minimizing the volume of HLW while reducing the radionuclide content of the LAW fraction to permit onsite disposal. Pretreatment includes the baseline process of enhanced sludge washing (Orme 1995).

4.1 BASELINE PRETREATMENT PROCESS

In this study, the baseline pretreatment process for evaluation of incremental separations of Hanford Site tank wastes includes the following:

- Previously separated radionuclides
- Solids/liquids separations, which includes enhanced (caustic) sludge washing.

The baseline pretreatment process removes an estimated total of 137.7 MCi of radionuclides from the TWRS processing inventory for offsite geologic disposal. The baseline separation includes 77.8 MCi of radionuclides that were originally separated and encapsulated (total original inventory), 6.38 MCi (HAPO shipments) that have been separated by other processing and are already shipped offsite, and 54 MCi of radionuclides (see Table 4-1) that will be separated with the HLW sludges (i.e., solids/liquids separations).

The baseline pretreatment processing results in an estimated total capital, operating, and geologic disposal cost savings of $9 billion (Colby 1995) over geologic disposal of all tank wastes (i.e., no solids/liquid separations).

4.1.1 Solids/Liquids Separations

The Tank Waste Remediation System (TWRS) process includes the characterizing, retrieving, treating, and disposing of chemicals contained in the SSTs and DSTs. Using simple separation techniques, the retrieved waste is segregated into a low-activity fraction containing the bulk of the non-radioactive constituents, and a high-activity fraction containing most of the solids. The bulk of the ^{90}Sr and TRU radionuclides are contained in the water insoluble fraction (i.e., solids) of the tank wastes. The bulk of the ^{137}Cs, ^{99}Tc, and essentially all of the ^{79}Se, ^{129}I, ^{14}C, and ^{3}H are contained in the soluble fraction of the tank wastes. The ^{79}Se, ^{129}I, ^{14}C, and ^{3}H are $\leq$ 0.01 MCi in the soluble fraction and essentially none exists in the insoluble phase. The processing inventory of radionuclides in the insoluble and soluble fractions was shown in Table 3-1. The insoluble fraction contains 55.6 of the 91.6 MCi projected to be in the tanks.

From the process flowsheet, batches of waste are retrieved into sludge wash tanks. Retrieved solids are washed four times using enough caustic (i.e., sodium hydroxide) to arrive at 3 molar NaOH solution and 8 wt% solids slurry in the product stream from caustic

washing. The caustic wash will solubilize additional amounts of aluminum, chromium, phosphate and sodium from the solids. Each wash is preceded by settling and decanting to remove liquids and concentrate the solids. Batches of leached and washed solids are combined and blended while the decanted liquids are combined for additional treatment.

The amount of radionuclides removed with liquid fraction is based on a solids carryover of 0.33 percent of the solids in the tank for each of three decants (Orme 1995). This results in an overall solids carryover of 1 percent for design purposes. To account for operational inefficiencies and uncertainties in solids/liquid separations, a more conservative solids carryover of 3 percent is applied for this evaluation. No further removal is provided for the 3 percent solids carryover. The liquids/solids separations process also assumes that the inefficiencies in washing interstitial supernatant from the solids results in carryover of 0.135 percent of the soluble radionuclides into the solids phase (Orme 1995).

The total of both inefficiencies during the solids/liquids separation process is shown in Table 4-1 and results in approximately 54 MCi in the solids slurry and 36 MCi in the liquid phase, with an additional 1.7 MCi entrained solids from an assumed 3 percent solids carryover. The 36 MCi in the separated liquids is considered for additional radionuclide separations in Section 4.2. No further separation is assumed for the 1.7 MCi entrained solids.

Table 4-1. Inventory For Additional Radionuclide Removals (MCi Decayed to December 31, 1999).

Radionuclide	Solids slurry fraction to high-level waste	Liquid fraction		Total
		Soluble	Entrained solids	
^{137}Cs	2.9	34.1	0.09	37.1
^{90}Sr	50.9	1.89	1.57	54.4
Transuranics	0.117	0.0096	0.004	0.131
^{99}Tc	0.0093	0.0228	-	0.032
^{79}Se	-	0.00103	-	0.00103
^{14}C	-	0.0053	-	0.0053
^{129}I	-	0.000051	-	0.000051
^{3}H	-	0.01	-	0.01
^{126}Sn	-	-	0.0016	0.0016
Uranium	0.00094	0.00006	-	0.001
Totals (rounded)	54	36	1.7	92

4.2 TECHNOLOGY OPTIONS

The NRC guidelines outlined in Chapter 2.0 have been used to assess radionuclide removal technologies and evaluate the practicality of achieving higher degrees of separation from the LAW fraction. This section focuses on the technologies for separation of radionuclides from supernatants to produce LAW. This evaluation consists of identifying: (1) individual technology options for radionuclide separations processes, (2) the status of the technology, (3) defining the radionuclide removal efficiency and (4) determining the cost of implementing the technology. The cost of implementing a given technology, with an estimated curie removal for the technology, is assessed in terms of cost per curie to provide a measure of economically practical. An economic assessment is provided only if a technology is considered to be technically practical.[2]

Technology options for radionuclide removal are considered for the radionuclides of interest as defined in Section 2.3.4. The radionuclides discussed in the sections below are ^{137}Cs, ^{90}Sr, TRU, ^{99}Tc, ^{79}Se, ^{14}C, ^{129}I, ^{3}H and uranium isotopes. The technology options for each of the key radionuclides are evaluated and compared using the definitions of technically and economically practical.

4.2.1 Cesium Removal

The liquids for pretreatment contain approximately 34.1 MCi cesium for further separations. Technology options considered in this study for removal of soluble cesium from the waste supernatants and wash liquors are: (1) a cesium ion exchange process using column operation, (2) in-tank precipitation or sorption process and, (3) volatilization of cesium from the LAW melter.

Cesium is the key radionuclide requiring additional removal to establish a new LAW limit. Cesium represents approximately 91 percent of the curies in the liquid fraction after solids/liquid separations have been performed. The distribution of cesium concentration in the tank wastes is shown in Figure 4-1. Adding the totals from the first three blocks in Figure 4-1 shows approximately 90 percent of the cesium curies is contained in 21 percent of the total volume.

[2]Cost estimates are based on 1995 dollars. In some cases, cost data were obtained from documents completed before 1995 and data are in 1993 or 1994 dollars. However, the overall costs will not be impacted by a 1 or 2 year cost escalation.

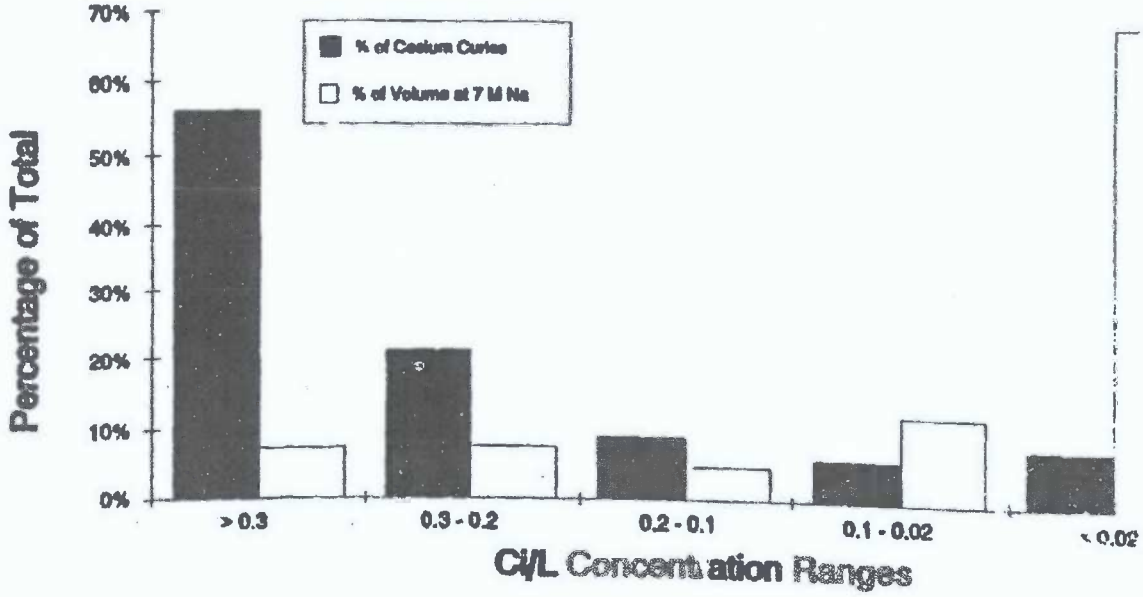


Figure 4-1. Waste Volume [illegible]

4.2.1.1 Ion Exchange—Column Operation. Cesium can be removed from alkaline solutions high in sodium with cation exchange resins. Beginning in 1967, cesium was extracted from tank wastes and recovered in B Plant using regenerable ion exchange processing. The use of regenerable cesium ion exchange in B Plant was the basis for radionuclide removal in the NRC determination. Ion-exchange using non-regenerable resins for cesium removal is also considered as a technology option.

The following sections discuss cesium ion exchange processes for single- and two-cycle technology options.

4.2.1.2 Single Cycle of Ion Exchange. The current design basis for evaluation of cesium ion exchange is based on ion exchange resin CS-100[3], and the laboratory data generated over several years of testing at Westinghouse Process Chemistry Laboratory and Pacific Northwest National Laboratory. It is an engineering judgement that the properties of CS-100 provide a conservative basis for ion exchange design. Alternate ion exchange materials, e.g., resorcinol formaldehyde, granular potassium cobalt hexacyanoferrate, crystalline silicotitanante (CST) and SuperLig™ 644[4], are also being evaluated for cesium removal from alkaline solutions (Lee et al. 1996). Silicotitanates have been used at the West Valley Demonstration Project (WVDP) facility at West Valley, New York for cesium removal from tank wastes. The silicotitanate used at West Valley was in the form of titania doped zeolite in columns.

The cesium ion exchange process in the reference flowsheet (Orme 1995) has a design basis of 99 percent removal of the soluble cesium from supernatants and wash liquors. The experience at B Plant with ion exchange during the cesium recovery campaigns for cesium capsule production was typically 90 percent recovery in order to maximize the total curies removed per unit time and not to minimize the cesium concentration in the treated liquid (Barton et al. 1986). A design basis of 99 percent cesium removal from the clarified supernatants and wash liquors is imposed as technically feasible. The cesium ion exchange process is evaluated based on a 97 percent cesium removal efficiency to allow for resin degradation and design and operational uncertainties.

Technical Practicality of Single-Cycle Ion Exchange. Cesium ion exchange with an organic resin was used at B Plant to recover the 55 MCi of ^{137}Cs for the encapsulation campaign at the Hanford Site. Removal of cesium using ion exchange technology is considered to be technically practical. Selection of an appropriate ion exchange material continues to be developed and tested.

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[3]Duolite CS-100 is a registered trademark of Rohm and Haas, Philadelphia, Pennsylvania.

[4]SuperLig 644 is a polymer resin of the covalently bound SuperLig™ macrocycle family of sequestering ligands from IBC Advanced Technologies (American Fork, Utah).

Cost of Single-Cycle Ion Exchange. The cost for cesium removal by ion exchange is based on data in the TWRS Environmental Impact Study (EIS) data package (Slaathaug 1995, Table F-36). The capital cost for cesium ion exchange is $380 million. The total operating cost for the cesium ion exchange system including decontamination and decommissioning (D&D) costs is $602 million to process all the waste. If all of the soluble waste is treated using a cesium ion exchange process, up to 33.1 MCi (97 percent separation efficiency) of ^{137}Cs is removed at a total cost of $982 million. This results in an average cost of $30/Ci ^{137}Cs.

The cost per incremental curie cesium removed is driven primarily by the operational costs of a cesium ion exchange facility. For this evaluation, a tank waste retrieval sequence was optimized to retrieve the tanks with the highest cesium concentration first. Data from the "baseline retrieval sequence" (Certa 1995) were optimized for removal of cesium. The retrieval sequence was prioritized to process only wastes for selected cesium concentrations. Output of the optimized sequence includes the waste volume processed and the resulting MCi cesium disposed in the LAW (Slaathaug 1996b). From this data, Figure 4-2 plots the $/Ci cesium removed versus the cesium concentration in the feed. For an optimized retrieval sequence, the operational cost per curie begins to increase significantly at concentrations less than 0.05 Ci ^{137}Cs/L. If a concentration of greater than 0.05 Ci ^{137}Cs/L is selected as the economically practical limit for terminating cesium removal processing, this results in a total cesium removal of 29 MCi cesium with approximately 5 MCi cesium remaining in the LAW.

Table 4-2. Incremental Operating Cost for Removing Additional Cesium.

Concentration of cesium for processing (Ci/L)	Optimized retrieval sequence			
	Million liters processed	MCi cesium in low-activity waste	Total cost in millions	Incremental operating cost ($/Ci)
<0.01	420.9*	1.0	982	147
0.01	255.7	1.9	850	59
0.02	130.5	2.6	749	28
0.05	88.7	4.8	715	11
0.1	78.1	5.6	706	5
0.2	59.6	8.6	691	3
0.3	29.6	15.7	667	2
0.5	6.5	23.6	648	0.5

*Total volume of decanted tank liquids, adjusted to 7*M* sodium, no wash solutions included.

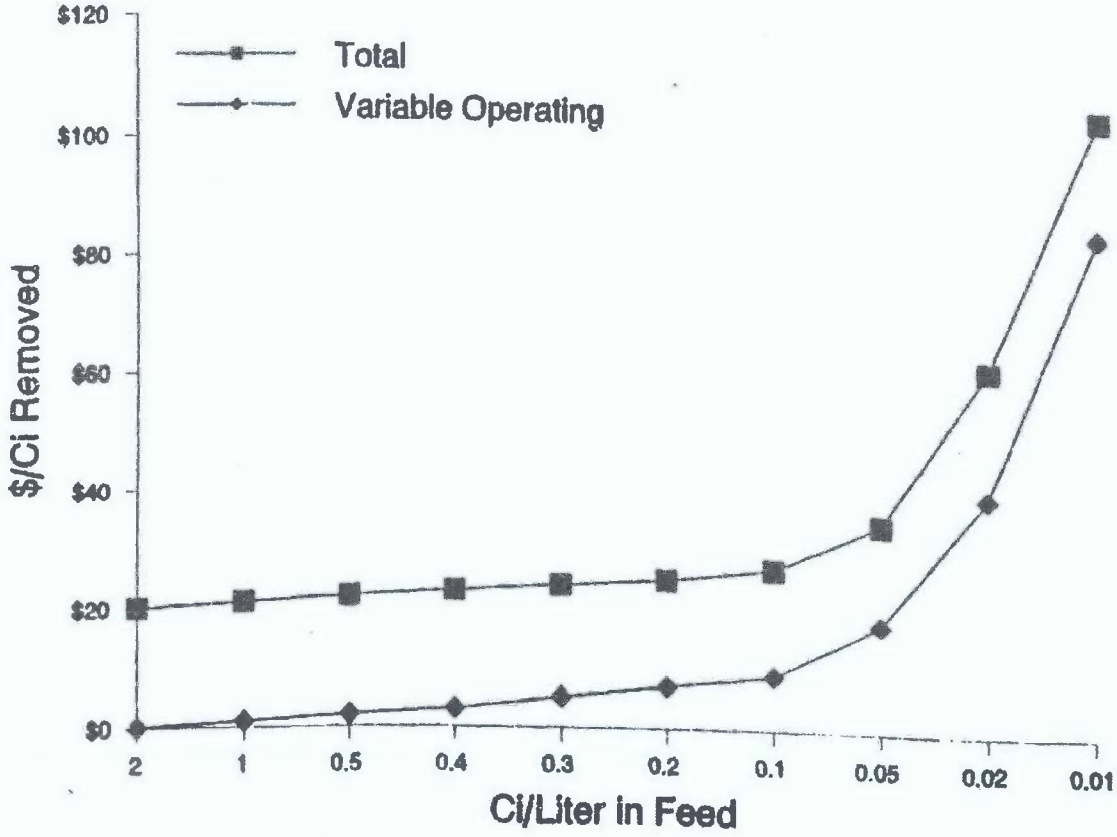

Figure 4-2. Cost per Curie Cesium Recovered versus Cesium Concentration in the Feed.

4.2.1.3 Second Cycle of Cesium Ion Exchange. A second cycle of cesium ion exchange treats the raffinate from the first ion exchange cycle and removes an additional 1.0 MCi ^{137}Cs assuming 97 percent recovery in the second cycle. This results in a total recovery of 99.9 percent for two cycles of cesium ion exchange and separates approximately 34 MCi cesium.

Technical Practicality of Second-Cycle Ion Exchange. The technical practicality of a second cycle of cesium ion exchange is similar to a single cycle of ion exchange.

Cost of Second-Cycle Ion Exchange. The costs associated with a second cycle of cesium ion exchange are derived from the difference in costs for separations facilities containing two cycle and one cycle of cesium ion exchange developed in the *Facility Configuration Report* (Boomer et al. 1993) (see Appendix B). The difference in capital costs attributed to the second cycle is $275 million. The operating cost differences for the second cycle of cesium ion exchange are $47 million for materials, $15 million for equipment replacement, and $83 million for D&D costs. If all soluble waste is treated, the additional 1 MCi of ^{137}Cs is separated for a cost of $420 million. This results in a cost of $420/Ci ^{137}Cs.

4.2.1.4 Cesium Precipitation In-Tank for All Wastes. Several materials have been proposed for removal of cesium from alkaline Hanford Site wastes by precipitation or sorption. These include nickel ferrocyanide, tetraphenyl borate and silicotitanates.

Sodium ferrocyanide and nickel sulfate were used at Hanford Site to remove cesium from SST supernatants. Concerns with (1) potential safety problems (nitrate-cyanide mixtures in the tanks), (2) a requirement to add acid to the wastes for pH adjustment before precipitation, and (3) process sensitivity to pH control have eliminated sodium ferrocyanide from consideration of full scale application for Hanford Site wastes.

The in-tank precipitation of cesium by the addition of tetraphenyl borate is used at the Savannah River Site (SRS) for treatment of tank wastes. Tetraphenylborate co-precipitates potassium with the cesium. Hanford Site wastes are different from the SRS wastes because Hanford Site wastes contain significant quantities of potassium. Hanford Site wastes contain 737 Mg of potassium, which would result in 6,750 Mg or 70 million L (18 Mgal) (d'Entremont and Walker 1987) of precipitate. The excessive quantity of chemical consumption and quantities of metal-organic precipitates to store and process through HLW vitrification have eliminated tetraphenyl borate from consideration.

Subsequent to the West Valley application of silicotitanates, research has evolved silicotitanates into a more selective material, CST. The CST materials have been developed by Sandia National Laboratories in conjunction with Texas A&M University and UOP Molecular Sieves as an alternative to organic ion exchange resins for cesium removal from Hanford Site tank wastes. The CST material has been manufactured into a suitable engineered form and is commercially available. If obtained in bulk quantities, CST may offer an increased ion exchange capacity, eliminate safety concerns regarding organic resin degradation, and reduce secondary waste generation compared to the use of organic resins.

CST is not regenerable. Therefore, the CST material cannot be eluted and must be disposed in a loaded form.

An evaluation of how the use of CSTs would impact HLW glass production was performed by Swenson (1995). The reference evaluated removal of ^{137}Cs to produce 1 Ci $^{137}Cs/m^3$ in the LAW and concludes that there is no benefit to in-tank Cs removal using CSTs. CST material contains TiO_2 which is predicted to have a low solubility in the HLW glass product (1 wt%). This results in a maximum usage of approximately 230 MT of TiO_2 before the 1 wt% TiO_2 is exceeded. Because of the large quantity of CST material required for an in-tank process for all waste, there is a significant increase of the HLW glass production whether by single-batch (14,303,900 MT at 1 percent titania) or column contacting (70,000 MT at 1 percent titania) of the CST with the waste. The current TWRS flowsheet (Orme 1995) projects 22,800 MT of HLW glass.

Technical Practicality of Cesium Precipitation In-Tank for All Wastes. Cesium precipitation in-tank for all wastes using CST is not practical due to the quantity of CST material required to remove greater than 99 percent of the cesium.

Cost of Cesium Precipitation In-Tank for All Wastes. No economic analysis is provided since the technology is not practical.

4.2.1.4.1 In-Tank Precipitation for Selected Wastes. Selective treatment of tank wastes is also an option for CST. Treating only the tank wastes more concentrated in cesium can significantly reduce CST usage and HLW volume. The selective CST treatment of tank wastes more concentrated in cesium can maximize the total cesium removed by a fixed amount of CST and the contained TiO_2. The HLW canisters (1.26 m^3 glass per canister is assumed for this report) produced are a function of the total titania contained in the CST used to treat all wastes. The quantity of CST used to treat the liquid phase of tank wastes is proportional to the volume of waste, degree of desired cesium removal [Decontamination Factor (DF)], and the method of supernatant/CST contact.

The selective CST treatment of tank wastes is based on maximizing the total cesium removed by a fixed amount of CST and the contained TiO_2. This optimization assumes that each batch of composite waste produced by the waste retrieval system is treated with a variable amount of CST and variable DF to produce a constant cesium to sodium ratio in the treated supernatants and an approximate constant cesium to TiO_2 ratio in the loaded CST.

The projected amount of vitrified HLW allows the addition of titania up to 1 percent. This results in a potential use of 230 MT of titania contained in CST without significant penalty, less than 136 canisters of additional HLW. The concentration of ^{137}Cs in waste tank supernatants also varies significantly between tanks.

An evaluation of selective treatments using CST is provided in Slaathaug 1996a. This study shows a 90 percent cesium removal is achieved with the incremental additions of CST equalling 230 MT TiO_2 (note: 230 MT TiO_2 is the amount allowable before exceeding the

1 wt% titania solubility limit in HLW glass) For single batch contracts, an estimated 8 MCi ^{137}Cs would remain in the LAW using this technology

Technical Practicality of In-Tank Precipitation for Selected Wastes. Selective treatment of tank wastes using CST is not technically practical for this analysis due to the development work still required (i.e. in-tank controls and scale-up issues). For this document, it is assumed further development is needed to accurately define the capacity of different forms of CST for application to Hanford Site tank wastes The reference study (Swenson 1995) recommends additional study to accurately define the capacity of different forms of CST and to increase the allowable concentration of titania in HLW glasses.

Cost of In-Tank Precipitation for Selected Wastes. Since the use of CST for cesium separation is not considered technically practical for this analysis, no economic assessment has been provided.[5]

4.2.1.5 Cesium Volatilization. The volatilization of some cesium during the melting of radioactive waste glasses is intrinsic to the vitrification process. A LAW melter could provide removal of volatile radionuclides from the LAW glass by routing volatile radionuclides captured in the melter offgas scrubber solution to the HLW feed stream instead of recycle to the LAW melter feed.

The melting of HLW glasses typically volatilizes 20 percent of the feed cesium from Joule heated melters. The HLW melters typically operate with a cold cap and a maximum temperature of 1100 °C to minimize cesium volatility. Operation of a Joule heated LAW melter at temperatures of 1300 °C or greater and with sparging to disrupt the cold cap could expect to increase cesium volatility up to 50 percent. Tests in FY 1995 (Higley 1995) with hot gas heated melters for LAW vitrification gave indicated cesium volatilities of 50 to 90 percent of the feed cesium.

Technical Practicality of Volatilization. Cesium volatilization and recovery is a design feature for full scale HLW melters. The details of process control needed to maximize and control cesium volatilization have not been established. The removal of cesium from aqueous off-gas treatment requires development and no production scale applications have been performed. Therefore, cesium removal using only a volatilization technology is considered not technically practical.

Cost of Volatilization. No economic analysis is provided since the technology is considered not technically practical.

[5]Appendix B, Cost Estimates, which contains limited cost data for CST that have been recently investigated, is provided for information only.

4.2.2 Strontium Removal

The liquid fraction for further pretreatment contain approximately 3.46 MCi ^{90}Sr (rounded to 3.5 MCi) for further separations. Of the 3.5 MCi, 1.89 MCi is soluble ^{90}Sr and 1.57 MCi (rounded to 1.6 MCi) is insoluble ^{90}Sr from solids carryover. No further removal is provided for the 1.57 MCi insoluble ^{90}Sr (see Section 4.1.1).

Technology options considered in this study for removal of soluble strontium from the waste supernatants and wash liquors are: (1) precipitation by chemical addition, (2) deactivation of the complexants, (3) complexant destruction and, (4) solvent extraction.

An analysis of DST radionuclide removal requirements on a tank-by-tank basis (Schulz 1995) has determined that no additional removal of soluble ^{90}Sr is required for any tank to meet the NRC Class C criteria. The ^{90}Sr concentration in glasses made from supernatant liquid in at least 21 of the DSTs will contain less than 150 Ci $^{90}Sr/m^3$ without pretreatment, which is below the Class B limit for ^{90}Sr. The highest concentrations of soluble strontium is found in CC wastes. Precipitation processes discussed for TRU removal result in ^{90}Sr that co-precipitates with the removal of soluble TRU (see Section 4.2.3.3).

The CC supernatant contains multivalent cations of radionuclides such as strontium and transuranics in greater concentrations than normal in alkaline solutions. This increased concentration occurs because of the presence of complexants added during previous strontium and cesium recovery campaigns in B Plant.

4.2.2.1 Precipitation of Strontium Phosphate from All Tank Wastes. Treating all the DST and SST wastes (177 tanks) at 5*M* Na to remove 1.8 MCi of the 1.9 MCi soluble ^{90}Sr would require 50,000 MT of $Sr(NO_3)_2$ and would increase the number of HLW canisters from 6,800 to 14,200.

The precipitation of strontium from the complexant bond by the addition of nonradioactive strontium involves isotopic exchange between two dissolved strontium atoms. This reaction may require rapid and intensive mixing before the nonradioactive strontium becomes unavailable for the exchange reaction due to precipitation as hydroxide or phosphate.

Technical Practicality of Strontium Phosphate Precipitation for All Tank Wastes. Precipitation of strontium phosphate has been investigated with limited hot bench scale testing and is considered not technically practical. Treating all the wastes would significantly impact (increase) the amount of HLW glass.

Cost of Strontium Phosphate Precipitation for All Tank Wastes. No economic analysis is provided since the technology is considered not technically practical at this time.

4.2.2.1.1 Precipitation of Strontium Phosphate from Selected Wastes. Recent radionuclide removal tests indicates that ^{90}Sr is effectively removed from CC waste (95 percent removal) by addition of 0.1 molar nonradioactive $Sr(NO_3)_2$ to precipitate $Sr_3(PO_4)_2$ (Schulz 1995).

Treatment of selected tanks of CC waste, 241-AN-102, 241-AN-107 and 241-AY-101, would remove 0.54 MCi of the soluble 0.57 MCi ^{90}Sr in these CC tanks. Treatment of the 10,800 m^3 in these three tanks would require the equivalent of 114 Mg of SrO be incorporated into the HLW glass. Assuming a HLW glass waste oxide loading of 45 wt% and an average SrO concentration of 0.5 wt% in blended HLW glass, an extra 75 canisters of HLW glass would result.

The precipitation of strontium from the complexant bound by the addition of nonradioactive strontium involves isotopic exchange between two dissolved strontium atoms. This reaction may require rapid and intensive mixing before the nonradioactive strontium becomes unavailable for the exchange reaction due to precipitation as hydroxide or phosphate.

Technical Practicality of Strontium Phosphate Precipitation for Selected Wastes. Precipitation of strontium phosphate has only been investigated with limited hot bench scale testing and is not technically practical.

Cost of Strontium Phosphate Precipitation for Selected Wastes. No economic analysis is provided since the technology is considered not technically practical at this time.

4.2.2.2 Strontium Removal by Complexant Destruction in All Wastes. The complexants that solubilize strontium in an alkaline solution can be destroyed in several minutes by thermal digestion at elevated temperature and pressure. A continuous flow application requires a temperature greater than 200 °C. After destruction of the complexants, the multivalent ions precipitate from the solution and are removed with solids/liquid separations equipment. The separated solids are slurried to DSTs for lag storage and future vitrification as HLW.

Technical Practicality of Complexant Destruction for All Wastes. The development status of complexant destruction by digestion at elevated temperature and pressure has been demonstrated by bench scale testing for commercial nonradioactive applications. No pilot plant testing or production scale processing has been performed on radioactive wastes similar to the Hanford Site tank wastes. Therefore, this process is considered not technically practical due to the required development work.

Cost of Complexant Destruction for All Wastes. No economic analysis is provided since the technology is not technically practical.

4.2.2.2.1 Strontium Removal by Complexant Deactivation for Selected Wastes. The complexants that solubilize strontium in an alkaline solution can be destroyed or deactivated by thermal digestion as discussed in Section 4.2.2.2. Application of thermal digestion for selected wastes can be performed in-tank. An analysis of in-tank heat and digest to deactivate the complexant in CC waste tanks 241-AN-102, 241-AN-107, and 241-AY-101 was performed by Klem (1995).

Technical Practicality of Complexant Deactivation for Selected Wastes. Testing of the heat and digest process is still in progress. The complexant is decomposed by slow chemical reactions performed by heating the tank contents to 90 °C to 100 °C for an extended period of time. The process is predicted to require 4.2 to 14.3 years to perform in-tank at 100 °C and the final performance is uncertain. Complexant deactivation for strontium removal is considered not technically practical.

Cost of Complexant Deactivation for Selected Wastes. No economic analysis is provided since the technology is considered not technically practical.

4.2.2.3 Strontium Removal by Solvent Extraction in All Wastes. The alkaline solutions that contain strontium are acidified using nitric acid. Strontium is removed from the acidified solution using a solvent extraction system similar to the processes described in the *Tank Waste Technical Options Report* (Boomer et al. 1993). Strontium recovery by solvent extraction is included with TRU separations. This process, using the solvent developed by Argonne National Laboratory (ANL), is called Strontium Extraction (SREX).

Technical Practicality of Solvent Extraction in All Wastes. Solvent extraction separation of strontium with SREX has been demonstrated in laboratory testing. However, no pilot plant testing or production scale processing using the SREX process has been performed on radioactive wastes similar to the Hanford Site tank wastes. Solvent extraction using other solvents has previously been used in B Plant at Hanford Site to recover strontium. Therefore, solvent extraction for strontium is considered technically practical.

Cost of Solvent Extraction in All Wastes. The cost for strontium removal by solvent extraction is based on facility designs presented in the *Facility Configuration Study* (Boomer et al. 1993). The cost of a solvent extraction system collocated with another processing facility is based on the cost differences between two pretreatment facilities with and without the solvent extraction system. These two facilities are identified as Solvent Extraction B (SOLEX B) and Sludge Wash B (SWB), with and without the solvent extraction, respectively, in the reference report (Boomer et al. 1993). The capital costs for these two facilities are updated for this report. A summary of the updated SWB cost estimate is found in Appendix B. The difference in costs between SOLEX B and SWB provides a basis for the cost difference of a LAW vitrification facility with and without solvent extraction capability. The difference in capital costs associated with transuranics removal by solvent extraction is $1.22 billion. The difference in operating costs associated with operation of the solvent extraction system is identified by the difference in operating costs for the SOLEX B and SWB facilities of $6.69 billion (Boomer et al. 1993).

The in-facility process for strontium removal by solvent extraction treatment of all wastes is expected to remove 3.4 MCi of the 3.5 MCi ^{90}Sr for further pretreatment. The removal of 3.4 MCi of ^{90}Sr at a total cost of $7.9 billion results in a radionuclide removal cost of $2,320/Ci of ^{90}Sr and is not economically practical. No economic analysis is currently available which provides the cost of solvent extraction for selective treatment of tank wastes (i.e., small scale processing).

4.2.3 Transuranic Removal

The liquid fraction for further pretreatment contains approximately 0.0096 MCi soluble TRU, rounded to 0.01 MCi, for further separation. Technology options considered in this study for removal of soluble TRU from the waste supernatants and wash liquors are: (1) complexant destruction, (2) deactivation of the complexants, (3) precipitation by chemical addition, and (4) solvent extraction. No further removal is provided for the 0.004 MCi of entrained, insoluble TRU fraction resulting from washing and decanting inefficiencies.

An analysis of DST radionuclide removal requirements on a tank-by-tank basis (Schulz 1995) has determined that only three DSTs (241-AN-102, 241-AN-107 and 241-AY-101) require some removal of soluble TRU from supernatants to meet the NRC Class C requirement of less than 100 nCi TRU/g glass. These three tanks are classified as CC wastes and contain an estimated 0.0054 MCi (56 percent) of the total soluble TRU.

The CC supernatant contains multivalent cations of radionuclides such as TRU and strontium in greater concentrations than normal in alkaline solutions. This increased concentration occurs because of the presence of complexants added during previous strontium and cesium recovery campaigns in B Plant.

4.2.3.1 Transuranics Removal by Complexant Destruction in All Wastes. The complexant deactivation of all wastes is discussed in Section 4.2.2.2 for removal of soluble strontium and is applicable to TRU removal. The complexant destruction process will remove an estimated 0.0099 MCi of the 0.01 MCi of soluble TRU in addition to the removal of the 1.87 MCi of soluble ^{90}Sr.

Technical Practicality of Complexant Destruction. As discussed in Section 4.2.2.2.1, TRU removal by complexant deactivation of all wastes is not technically practical.

Cost of Complexant Destruction. No economic analysis is provided since the technology is not technically practical.

4.2.3.1.1 Transuranic Removal by Complexant Deactivation in Selected Wastes. The complexant deactivation of selected CC wastes (241-AN-102, 241-AN-107, and 241-AY-101) was discussed in Section 4.2.2.2.1 for removal of soluble strontium and is applicable to TRU removal. The complexant deactivation process will remove an estimated 0.005 MCi of the 0.0054 MCi of the soluble TRU in the CC wastes.

Technical Practicality of Complexant Deactivation in Selected Wastes. As discussed in Section 4.2.2.2.1 for strontium removal, complexant deactivation for TRU removal is considered not technically practical.

Cost of Complexant Deactivation in Selected Wastes. No economic analysis is provided since the technology is not technically practical.

4.2.3.2 Transuranic Removal From All Wastes by Solvent Extraction. The solutions that contain transuranics in an alkaline solution are acidified using nitric acid. Transuranics are removed from the acidified solution using a solvent extraction system similar to the processes described in the *Tank Waste Technical Options Report* (Boomer et al. 1993). Transuranic recovery by solvent extraction is part of Solvent Extraction B. The transuranic extraction process is generally called TRUEX.

Technical Practicality of Solvent Extraction for All Wastes. Solvent extraction separation using TRUEX has been demonstrated through laboratory testing and pilot scale testing. Solvent extraction using other solvent extraction processes has previously been used in Reduction Oxidation (REDOX) and Plutonium-Uranium Extraction (PUREX) at the Hanford Site and at other sites to recover uranium, plutonium and neptunium. Transuranics removal by TRUEX is technically practical based on the extensive laboratory and pilot scale tests.

Cost of Solvent Extraction for All Wastes. The cost for transuranic removal by solvent extraction is based on facility designs presented in the *Facility Configuration Study* (Boomer et al. 1993). The cost of a solvent extraction system collocated with another processing facility is based on the cost differences between two pretreatment facilities with and without the solvent extraction system. These two facilities are identified as SOLEX B and SWB, with and without the solvent extraction, respectively, in the reference report (Boomer et al. 1993). The capital costs for these two facilities are updated for this report. A summary of the updated SWB cost estimate is found in Appendix B. The difference in costs between SOLEX B and SWB provides a basis for the cost difference of a LAW vitrification facility with and without solvent extraction capability. The difference in capital costs associated with transuranics removal by solvent extraction is $1.22 billion. The difference in operating costs associated with operation of the solvent extraction system is identified by the difference in operating costs for the SOLEX B and SWB facilities of $6.69 billion (Boomer et al. 1993).

The solvent extraction treatment of wastes is expected to remove 0.01 MCi of the transuranics. The removal of 0.01 MCi transuranics at a total cost of $7.9 billion results in a radionuclide removal cost of more than $790,000/Ci of transuranics and is not economically practical.

4.2.3.3 Hydroxide Precipitation of Transuranics from Selected Wastes. The analysis of DST radionuclide removal requirements (Schulz 1995) indicates that TRU removal for the three CC waste DSTs (241-AN-102, 241-AN-107 and 241-AY-101) can be attained with additions of NaOH as demonstrated by laboratory testing (Herting 1993, 1994a, 1994b, Washington 1990). The results indicate that wastes in tanks 241-AN-107 and 241-AY-101 can produce < 100 nCi TRU/g LAW glass by adjustment of the supernatants to 3.0*M* and 1.5*M* hydroxide ion, respectively, and addition of 0.01*M* ferric ion to serve as a $Fe(OH)_3$ carrier precipitate. Data for hydroxide adjustment for 241-AN-102 are not available at this time. The available laboratory data indicate that TRU removal increases with additional contact time. The development work also indicates that approximately 20 percent of the soluble strontium may be co-precipitated by the hydroxide precipitation process (0.1 MCi of the 0.57 MCi soluble ^{90}Sr in the three CC tanks).

Laboratory testing was also performed using 0.1*M* $Fe(NO_3)_3$ and 3.0*M* NaOH for the precipitation of TRU. The TRU is adsorbed on the surface of the ferric hydroxide floc formed by precipitation. In-tank mixers are required to keep the floc in suspension and available to the soluble TRU for adsorption. As demonstrated from laboratory testing, precipitation of $Fe_2O_3.xH_2O$ from CC waste removes 84 to 96 percent of the ^{241}Am and 60 to 75 percent of the soluble plutonium (Schulz 1995). The development work also indicates that approximately 50 percent of the soluble strontium may also be removed by the ferric hydroxide precipitation process (0.28 MCi of the 0.57 MCi soluble ^{90}Sr). However, the addition of iron impacts the volume of HLW glass produced.

Technical Practicality of Hydroxide Precipitation for Selected Wastes. It has been determined (Schulz 1995) that only three DSTs require some removal of soluble TRU from supernatants. Laboratory testing has been performed with waste from two of the three CC tanks and in-tank mixing with chemical addition has been well-demonstrated. Therefore, this technology is considered to be technically practical. For three CC tanks, containing an estimated 0.0054 MCi soluble TRU, 0.004 MCi is removed using hydroxide precipitation. The quantity of co-precipitated ^{90}Sr for this technology (0.1 MCi) is subtracted from the total ^{90}Sr to LAW.

Cost of Hydroxide Precipitation for Selected Wastes. The total costs for adjustment of the hydroxide concentration and addition of 0.01*M* iron in the DSTs (several weeks or months before retrieval and solids/liquid separation) is $6 million to $18 million (Appendix B provides additional cost data).

The addition of 19*M* NaOH to the three tanks to obtain a final concentration of 3*M* NaOH increases the total tank waste sodium by 250 MT or 0.14 equivalent vaults. The disposal cost per vault of solidified LAW is approximately $10 million (Slaathaug 1995, Table F-36) and the incremental Na usage results in $1.4 million increase in LAW disposal costs. The addition of NaOH to the tanks is assumed to cost $0.6 million. Therefore, the LAW cost increase is approximately $2 million.

The addition of the 0.01*M* iron results in an additional 5 to 21 canisters of HLW glass. The canisters are assumed to contain 1.26 m^3 of glass and a maximum Fe_2O_3 content of 12 wt% (Orme 1994). The minimum HLW canister impact is based on 50 wt% Fe_2O_3 if the HLW facility is processing glass at less than the 12 wt% Fe_2O_3 limit and the maximum HLW canister impact is based on the 12 wt% Fe_2O_3 limit. With an incremental HLW canister cost of $745,000 per HLW canister (Slaathaug 1995), this results in an additional operating cost of approximately $3.7 to $16 million.

The resulting $6 million to $18 million total cost for removal of approximately 0.004 MCi TRU (from three tanks) necessary to meet the < 100 nCi/g criteria results in a cost of $1,500 to $4,500/Ci TRU. If the 0.1 MCi of co-precipitated ^{90}Sr is included, the cost is reduced to $60/Ci to $170/Ci.

Cost of Ferric Hydroxide Precipitation for Selected Wastes. The addition of 0.1*M* $Fe(NO_3)_3$ and 3.0*M* NaOH for precipitation of TRU to the three CC tanks results in a maximum additional 50 to 210 canisters of HLW glass. The canisters are assumed to contain 1.26 m^3 of glass and a maximum Fe_2O_3 content of 12 wt% (Orme 1994). The minimum HLW canister impact is based on 50 wt% Fe_2O_3 if the HLW facility is processing glass at less than the 12 wt% Fe_2O_3 limit and the maximum HLW canister impact is based on the 12 wt% Fe_2O_3 limit. With an incremental HLW canister cost of $745,000 per HLW canister (Slaathaug 1995), this results in an additional operating cost of approximately $37 to $160 million. The incremental LAW costs are $2 million due to the additional sodium (see economics of hydroxide precipitation) and provides a $39 million to $162 million total cost for removal of approximately 0.005 MCi TRU and 0.28 MCi co-precipitated ^{90}Sr. This results in a removal cost of $140/Ci to $570/Ci.

4.2.3.4 Transuranics Removal by Complexant Deactivation in Selected Wastes. The complexant deactivation of selected CC wastes (241-AN-102, 241-AN-107 and 241-AY-101) is discussed in Section 4.2.2.2.1 for removal of soluble strontium. An analysis of [illegible] tank heat and digest to deactivate the complexant in CC waste tanks was performed by Klem (1995). The complexant deactivation process will remove an estimated 0.005 MCi of the 0.0054 MCi of the soluble TRU in addition to the removal of the 0.54 MCi of soluble ^{90}Sr.

Technical Practicality of Complexant Deactivation for Selected Wastes. The complexant deactivation process for CC wastes is expected to require 4.2 to 14.3 years in-tank at 100 °C and the final performance is uncertain. Therefore, complexant deactivation is not technically practical.

Cost of Complexant Deactivation for Selected Wastes. No economic analysis is provided since the technology is not technically practical.

4.2.4 Technetium-99 Removal

The soluble ^{99}Tc fraction of the tank waste contains an estimated 0.0228 MCi. Incorporating the entire quantity of technetium (0.032 MCi, soluble and insoluble) into the LLW glass would result in approximately 0.2 Ci $^{99}Tc/m^3$ glass. On a tank-by-tank basis, no additional removal of ^{99}Tc is required to meet the Class C limit of 3 Ci/m^3 in the solidified LAW (Schulz 1995). However, ^{99}Tc removal may be necessary to meet the disposal system requirements based on the performance assessment.

No demonstrated technology exists for in-tank precipitation of soluble ^{99}Tc from DST waste solutions. Candidate precipitation compounds include Tc_2S_7 and tetraphenylphosphonium pertechnetate (Schulz et al. 1995). Candidate technology options for removal of soluble technetium in the pretreatment or vitrification facility include mixed bed ion exchange, anion resin ion exchange, and volatilization.

4.2.4.1 Mixed Bed Technetium and Cesium Ion Exchange. Argonne National Laboratory and Eichrom Industries, Inc., have developed a technetium-specific resin for use in laboratory and analytical applications (Horwitz 1995). Preliminary scoping studies of applying resin to large scale processing of Hanford Site tank wastes has been performed. The resin is tailored on the molecular level to specifically absorb technetium. The resin, named Tc·Spec 5, is commercially available as an analytical reagent. Tc·Spec 5 strongly absorbs technetium from solutions greater than 4*M* salt and technetium elutes with water or dilute nitric acid. These properties are identical to the chemical behavior of cation ion exchange resins used for cesium removal. Operation of a single ion exchange column system containing a mixture of the resins allows simultaneous recovery of technetium and cesium. A mixed bed ion exchange system is a common industrial practice and is significantly less expensive than building and operating two ion exchange systems in series.

The laboratory work to date for the application to Hanford Site tank wastes is initial proof of principal experiments with 241-SY-101 simulant only to determine distribution coefficients. Use of Tc·Spec 5 in a mixed bed configuration eliminates the use of a water scrub to displace interstitial feed from the column before cesium elution. The water displacement flush would prematurely elute technetium. The water flush would have to be replaced with draining the interstitial feed from the column. The less efficient removal of feed sodium from the column by draining would have a currently undefined impact by increasing HLW volume. If the increase in HLW volume is excessive, alternate designs with Tc·Spec 5 columns in series with the cesium ion exchange columns would be investigated. Significant additional cold and hot laboratory development work is required.

Technical Practicality of Mixed Bed Ion Exchange. The mixed bed ion exchange recovery of ^{99}Tc with Tc·Spec is not technically practical at this time.

Cost of Mixed Bed Ion Exchange. No economic analysis is provided since the technology is not technically practical.

4.2.4.2 Technetium Anion Exchange. Technetium can be removed from alkaline solutions high in sodium with anion exchange resins. The removal of technetium with a strong base anion exchange resin has been demonstrated with a single large scale batch loading of the resin at the Hanford Site (Beard and Caudill 1964). The loaded resin was shipped to another DOE site for elution. Multiple load and elution cycles of technetium on ion exchange resins has not been demonstrated.

The current design basis for evaluation of technetium ion exchange is based on a strong-base organic anion exchange resin (Schultz 1980). More recent analyses have been performed on the sorption of technetium from DSSF simulants using Reillex-HPQ anion exchange resin and is limited to hot lab scale experiments.

The technetium ion exchange process loads technetium from the alkaline feed stream onto the resin by passing the feed through a column of resin particles. When breakthrough of technetium occurs in the column raffinate, the feed is diverted to a parallel column and the loaded column is flushed with water. Following the flushing operation, the technetium is

eluted from the column with strong nitric acid (6*M* HNO_3). Following the technetium elution, the column is flushed with water. The column is then ready for another feed loading cycle.

The technetium product stream eluted from the column with nitric acid is concentrated and the bulk of the nitric acid boiled off from the technetium solution for fractionation and recycle to the next elution cycle. The residual nitric acid in the concentrated technetium product is then neutralized with sodium hydroxide. The neutralized technetium concentrate is then transferred to a DST for interim storage before vitrification as HLW.

Technical Practicality of Anion Exchange The technetium ion exchange process meets the definition of being technically practical based on the large demonstration recovery of 27 Ci of ^{99}Tc at the Hanford Site using technetium ion exchange (Beard and Caudill 1964). However, information contained in Schroeder et al. (1995) indicate some serious questions for application of ion exchange to existing Hanford Site tank wastes due to the uncertainties in the valence state of the technetium (i.e., the pertechnate ion, valance 7, is extractable but other oxidation states are not extractable.) Therefore, ^{99}Tc anion exchange is not considered technically practical at this time.

Cost of Anion Exchange. The cost for technetium removal by ion exchange is based on facility and equipment designs presented in the *Tank Waste Technical Options Report* (Boomer et al. 1993). The cost of a technetium ion exchange system collocated with another processing facility is based on the cost differences between two pretreatment facilities with and without the technetium ion exchange process. These two facilities are identified as Sludge Wash D (SWD) and Sludge Wash B (SWB), with and without the technetium ion exchange, respectively, in the reference report (Boomer et al. 1993). Both facilities include sludge washing, solids/liquids separation, and cesium ion exchange. The capital costs for these two facilities are updated for this report. Summaries of the updated SWD and SWB cost estimates are found in Appendix B. The difference in costs between SWD and SWB provides a basis for the cost difference of a LAW treatment or vitrification facility with and without technetium ion exchange capability. The difference in capital costs associated with technetium ion exchange processing is $260 million. The difference in operating costs associated with operation of the technetium ion exchange processing is identified by the difference in operating costs for the SWD and SWB facilities of $400 million (Boomer et al. 1993).

The technetium ion exchange process recovers 0.0226 MCi of ^{99}Tc for a total cost of $660 million. This results in a cost of $29,000/Ci ^{99}Tc and is not economically practical.

4.2.4.3 Technetium Volatilization. The volatilization of technetium from glass melters and calciners is an intrinsic feature of the operating temperatures. Processes and melters for the vitrification of HLW glass are designed to suppress volatilization and recycle volatilized technetium to the melter. Design of the LAW melter features, control of melter chemistry, and removal of technetium from the offgas stream with minimal recycle to the melter results in the LAW melter system providing technetium removal. The degree of removal that can be obtained by this process needs to be confirmed by additional development.

Reviews of technetium volatility from glasses have shown up to 95 percent removal (DF of 20) of technetium (Langowski 1995, Vida 1994). Melter design can be modified to enhance technetium volatility by disruption of the cold cap, sparging the molten glass, or use of a gas heated melter. The melter chemistry can be modified to enhance technetium volatility by increased temperature and/or changing from reducing to oxidizing conditions.

The addition of technetium removal to a volatilization system requires neutralization with sodium hydroxide and addition of sodium sulfide to the scrubber concentrate for technetium precipitation as Tc_2S_7. The FY 1995 flowsheet for TWRS Optimized Processing Strategy (Slaathaug 1996a) shows NaOH and Na_2S are added to the concentrate and before filtration for separation of the technetium sulfide. The technetium depleted filtrate is recycled to the melter feed tank. The Tc_2S_7 is slurried to a DST for interim storage before HLW vitrification. An alternate method of ^{99}Tc removal from the concentrate is to use absorption on Tc Spec 5 in a column.

Technical Practicality of Volatilization. The technetium volatilization process is not technically practical because the process has not been adequately demonstrated.

Cost of Volatilization. No economic analysis is provided since the technology is not technically practical.

4.2.5 Selenium-79 Removal

The tank waste contains an estimated 0.00103 MCi of ^{79}Se. All of the selenium in the tank waste is assumed to be soluble.

No demonstrated technology exists for in-tank precipitation of soluble ^{79}Se from DST waste solutions. In-tank precipitation would allow selenium to be removed by solids/liquids separations process discussed in Section 4.1.1. Although selenium is known to precipitate from water upon addition of sulfur dioxide, it is not known whether this technique would work as an in-tank precipitation process due to the very small amount of selenium present in the waste. A cursory review of commercial selenium recovery processes has not revealed any processes suitable for direct removal of selenium from the Hanford Site tank waste solutions during pretreatment. Laboratory methods for removal of selenium have not been reviewed for scaled up application in a pretreatment or vitrification operation. Ion exchange processes may exist, but the identification of selenium specific ion exchange resin has not been pursued.

4.2.5.1 **Selenium Volatilization.** The volatilization of selenium from commercial glass melters and the pyrometallurgical industry is a well known phenomena. Selenium is commercially recovered by the roasting of copper ores and residues from electrolytic copper recovery. Selenium is recovered from roasting flue deposits by sublimation or leaching.

The commercial glass industry has routinely experienced selenium losses of up to 80 percent or more (Kirk-Othmer 1982). Melter design can be modified to enhance selenium volatility by disruption of the cold cap, sparging the molten glass, or use of a gas heated melter. The melter chemistry can be modified to enhance selenium volatility by increased temperature and/or by changing from reducing to oxidizing conditions. Melter design could also include removal of selenium from the offgas stream with no recycle to the melter.

An option for selenium purification involves precipitation with sulfur dioxide in the scrubber solution from a LAW melter and offgas system. It is assumed that selenium can be co-precipitated with technetium by neutralization of the scrubber concentrate with sodium hydroxide and addition of sodium sulfide to precipitate selenium as a sludge (Kirk-Othmer 1982). The selenium filtrate can be recycled to the melter feed tank. The selenium sludge and Tc_2S_7 is slurried to a DST for interim storage before HLW vitrification.

Technical Practicality of Selenium Volatilization. The selenium volatilization process is not technically practical since the process has not been demonstrated at plant scale for radioactive wastes.

Cost of Selenium Volatilization. No economic analysis is provided since the technology is not technically practical.

4.2.6 Carbon-14 Removal

The estimated inventory of ^{14}C in the tank wastes is 0.0053 MCi ^{14}C. The estimated inventory of 0.0053 MCi ^{14}C represents 120 kg of the ^{14}C isotope and is diluted by approximately 1,800,000 kg of natural carbon in the tank wastes as carbonate and organics. The chemistry of carbon results in ^{14}C being distributed in supernatants and solids of all tanks. The organic carbon and carbonate content of the wastes will be converted to carbon dioxide, CO_2, as a result of the LAW vitrification process. In the vitrification process a total of 6,500,000 kg of CO_2 will be released in the offgas. To capture and remove the ^{14}C, the 6,500,000 kg of CO_2 could be absorbed in slaked lime as 15,000,000 kg of $CaCO_3$.

An analysis of the impact of not removing ^{14}C and ^{129}I indicates the maximum offsite individual will receive a 50-year dose commitment from atmospheric releases of less than 0.7 mrem/yr (Colby 1994). This is 0.2 percent of the national average individual dose from background of 300 mrem/yr.

Technical Practicality of Carbon Removal. There is no currently known isotopic enrichment technology that can separate the ^{14}C isotope from natural ^{12}C for Hanford Site tank wastes. Therefore, ^{14}C separation for disposal as HLW is not technically practical.

Cost of Carbon Removal. No economic analysis is provided since the technology is not technically practical.

4.2.7 Iodine-129 Removal

The estimated inventory of ^{129}I in the tank wastes is 0.000051 MCi (51 Ci). Iodine confinement technology that has been developed in the nuclear industry was aimed at relatively high concentrations and purity as released in reactor accidents or from nuclear fuel reprocessing dissolvers. The relatively high concentrations and absence of organics allows evolution of the resulting iodine compounds into the vapor phase and sorption with solids or liquids. The absence of chlorides and fluorides also eliminates competitive coabsorption of halides on solid sorbants. However, the alkaline tank wastes contain significant quantities of halides and organic materials.

Joule heated melters used for vitrification of the LAW typically operate under reducing conditions to protect the electrodes from catastrophic oxidation. Operation under reducing conditions will result in significant organic content in the melter offgas. The technology for organic iodide confinement in reprocessing was never fully satisfactory, and efforts were made to limit organic input into the dissolver system (the amount of PUREX organic diluent soluble in recycle water from the acid fractionator was of concern).

The iodine concentration in the tank wastes is typically 1,000 to 10,000 times lower than would exist in commercial fuel dissolver solutions that iodine removal technology was developed for. Thus, there is no technically practical iodine removal and confinement technology for tank wastes.

The path of ^{129}I in the LAW vitrification process is release to the atmosphere and an unknown quantity to the chloride purge stream. The chemistry of iodine suggests the potential removal of iodine from the chloride purge stream in the LAW offgas treatment system. The similar chemical behavior of the halides (fluorine, chlorine, and iodine) suggest that significant quantities, 20 to 80 percent, of the ^{129}I inventory may accumulate with the chloride and fluoride streams concentrated in the offgas treatment system for purge and disposal as grout. If ^{129}I concentrates in these streams, technology could be investigated to determine if cost effective systems for separation of ^{129}I from mixed chlorides and fluorides exist.

An analysis of the impact of not removing ^{14}C and ^{129}I indicates that if all is released to the atmosphere, the maximum offsite individual will receive a 50-year dose commitment from atmospheric releases of less than 0.7 mrem/yr (Colby 1994). This is 0.2 percent of the national average individual dose from background of 300 mrem/yr.

Technical Practicality of Iodine Removal. ^{129}I removal is considered not technically practical because no technology has been demonstrated for the relatively small concentrations in the Hanford Site tank wastes.

Cost of Iodine Removal. No economic analysis is provided since no ^{129}I removal technology has been provided.

4.2.8 Tritium Removal

Tritium (^3H) contained in the tank wastes is estimated to be 10,000 Ci (Colby 1994) and will be discharged from the pretreatment and waste vitrification facilities in the process condensates as tritiated water. The excess process condensates that are not recycled within the TWRS processing operation are routed to the 200 Area Effluent Treatment Facility (ETF).

The ETF process does not remove the tritium isotope from the natural hydrogen in the water. The separation of tritium from the treated condensate was evaluated and it was determined not to be technically practical (DOE 1994b).

The chosen alternative for disposal of the tank waste tritium in the treated water is to discharge this water to the subsurface and allow tritium to decay into non-radioactive helium before it reaches the Columbia River. The effluent infiltration gallery, also known as the State Approved Land Disposal Site (SALDS) is located just north of the 200 West Area. A study (Golder Associates 1990) shows that SALDS provides a 105-year travel time before tritium bearing ground water discharges to the Columbia River. The study concludes that through natural decay and subsurface dispersion, the concentration of tritium will be well within drinking water standards when it reaches the Columbia River.

4.2.9 Tin Removal

The tank waste for pretreatment contains an estimated 0.0016 MCi of insoluble ^{126}Sn. No technologies for tin removal were reviewed for this evaluation. For performance assessment studies, some tin to the LAW fraction is assumed to ensure continued consideration of tin for intruder dose consequences. The ^{126}Sn is not a significant contributor to the total curies and, as discussed for other radionuclides with small inventories, would not be economically practical for removal.

4.2.10 Uranium Removal By Solvent Extraction for All Wastes.

The soluble waste fraction contains an estimated 0.00006 MCi (60 Ci) of uranium isotopes. Uranium was routinely recovered and shipped as a product at the Hanford Site and other nuclear reprocessing facilities around the world. The technology most applied in the past for uranium is solvent extraction with tributyl phosphate in a hydrocarbon diluent. The solvent extraction recovery of uranium from the alkaline supernatants and dissolved salt cakes requires acidification of the supernatants and dissolved salts before the solvent extraction process. The acidification of tank wastes was demonstrated by the recovery of both uranium and strontium from Hanford Site tank wastes followed by solvent extraction recovery in U Plant and B Plant, respectively.

The alkaline solutions that contain uranium are acidified using nitric acid. Uranium is removed from the acidified solution using a solvent extraction system similar to the processes described in the *Tank Waste Technical Options Report* (Boomer et al. 1993). The process is generally called TRUEX. Uranium is recovered by solvent extraction only when the TRUEX process is available and used after the supernatant is acidified. Since uranium exhibits alkaline solubility and is distributed over all the wastes, all wastes are assumed for treatment.

Technical Practicality of Solvent Extraction. Solvent extraction using c r solvent extraction processes has previously been used in U Plant, REDOX and PU X at the Hanford Site to recover uranium, plutonium and neptunium. No production scale processing using the TRUEX process has been performed on radioactive wastes similar to the Hanford Site tank wastes. However, uranium removal by solvent extraction separation using TRUEX is technically practical based on the extensive laboratory and pilot scale tests.

Cost of Solvent Extraction. The cost for uranium removal by solvent extraction is based on facility designs presented in the *Tank Waste Technical Options Report* (Boomer et al. 1993). The cost of a solvent extraction system collocated with another processing facility is based on the cost differences between two pretreatment facilities with and without the solvent extraction system. These two facilities are identified as Sludge Wash B (SWB) and Solvent Extraction B (SOLEX B), with and without the solvent extraction, respectively, in the reference report (Boomer et al. 1993). The capital costs for these two facilities have been updated for this report (see Appendix B for cost estimates). The difference in costs between SOLEX B and SWB provides a basis for the cost difference of a LAW vitrification facility with and without solvent extraction capability. The difference in capital costs associated with solvent extraction is $1.22 billion. The difference in operating costs associated with operation of the solvent extraction system is identified by the difference in operating costs for the SOLEX B and SWB facilities of $6.69 billion (Boomer et al. 1993).

The solvent extraction treatment of all wastes is expected to remove 60 Ci of the soluble uranium. Combining the recovery of 3.4 MCi soluble strontium, 0.0095 MCi transuranics and 60 Ci uranium results in a potential total recovery of 3.4 MCi for about $7.9 billion and results in a removal cost of approximately $2,320/Ci. Solvent extraction for all wastes is considered not economically practical.

4.2.11 Radionuclide Removal Technology Options Summary

The radionuclide removal technology options discussed in the sections above are summarized in Tables 4-3 and 4-4. Table 4-3 addresses the "technically practical" criteria and summarizes the technology status of options discussed. Table 4-4 provides the costs and the MCi separated for the technologies determined to be technically practical.

Table 4-3. Summary Table of Technology Maturity Assessment.

Technology option for Hanford Site tank waste	Technically practical (Yes/No)	Concept	Cold bench scale studies	Bench scale demonstration		Pilot scale demonstration		Production scale use	
				Cold	Hot	Cold	Hot	Conventional industry	Nuclear industry
Core - Previous Separations, Solids/Liquids Separations	Yes								
Single-Cycle Cation Ion Exchange for ^{137}Cs	Yes								
Second-Cycle Cation Ion Exchange for ^{137}Cs	Yes								
In-Tank Precipitation for Cesium, All Wastes	No								
In-Tank Precipitation for Cesium, Selective Treatment	No								
Volatilization	No[a]								
Strontium Phosphate Precipitation, $Sr_3(PO_4)_2$	No								
Complexant Destruction for ^{90}Sr and TRU in All Wastes	No								
Selective Complexant Deactivation for ^{90}Sr and TRU, Selected Wastes	No								
Hydroxide Precipitation for TRU (and co-precipitated ^{90}Sr) (Includes ferric hydroxide)	Yes[b]								
Mixed Bed Ion Exchange (^{99}Tc and ^{137}Cs)	No								
Anion Resin Ion Exchange (^{99}Tc)	No								
Solvent extraction: SREX	No								
Solvent Extraction: TRUEX	Yes								
Solvent extraction: PUREX	Yes								

Shaded area = Demonstrated
CST = crystalline silicotitanate
PUREX = Plutonium-uranium extraction
TRUEX = Transuranic extraction
SREX = Strontium extraction

[a]Volatilization as a technology for cesium removal is not considered to be technically practical. However, some radionuclide volatilization occurs intrinsic to the melter operation.

[b]Technology maturity is assessed based on laboratory testing with actual tank wastes from two of the three tanks requiring TRU removals.

Table 4-4. Summary Table of Costs for Technically Practical Radionuclide Removal Technology Options.

Technically practical technology option	Economically practical (Yes/No)	Cost, $ millions			MCi Removed[a]				Cost, $/Ci
		Capital	Operating	Total	^{90}Sr	^{137}Cs	TRU	^{99}Tc	
Baseline Process - Previous Separations, Solids/Liquids Separations		-	-	-	77.5	60.2	0.120	0.009	-
Single-Cycle Cation Ion Exchange, Selective Removal (cesium concentrations >0.05 Ci/L)	Yes	380	335	715	-	29	-		25
Single-Cycle Cation Ion Exchange, Selective Removal (cesium concentrations <0.05 Ci/L)	No	-	267	267	-	4.1	-	-	65
Single Cycle Cation Ion Exchange	No	380	602	982	-	33.1	-		30
Second-Cycle Cation Ion Exchange	No	275	145	420	-	1.0	-		420
Hydroxide Precipitation for TRU and ^{90}Sr, Selective Treatment	Yes	0	6 - 18	6 - 18	0.1	-	0.004		[illegible]
Ferric Hydroxide Precipitation for TRU and ^{90}Sr, Selective Treatment	No	0	39 - 162	39 - 162	0.28	-	0.005		[illegible]
Solvent Extraction, TRUEX, PUREX	No	1220	6690	7900	-	-	0.0095		[illegible]

PUREX = Plutonium-uranium extraction
TRU = Transuranic
TRUEX = Transuranic extraction
[a] ^{14}C, ^{129}I, ^{3}H, ^{79}Se and uranium are excluded from Table 4-4 due to insignificant quantities for removal, are minor contributors to total activity, and/or have no practical removal technology demonstrated.

5.0 RADIONUCLIDE SEPARATIONS TO MEET THE U.S. NUCLEAR REGULATORY COMMISSION GUIDELINES

The goal of the LAW determination is to establish an inventory for all key radionuclides in the LAW that satisfies the regulatory intent of "technically and economically practical" and meets the limits and performance objectives of 10 CFR Part 61. This section summarizes the evaluation of the three NRC guidelines.

5.1 REMOVE RADIONUCLIDES TO THE MAXIMUM EXTENT PRACTICAL

Radionuclide removal technologies evaluated to meet the removal to the maximum extent practical guideline include previous removals, solids/liquids separation, and additional removal from the liquid fraction. Table 5-1 shows the radionuclide removal technologies and curies removed that are identified as both technically and economically practical. Additional removal technologies from the liquid fraction include selective TRU removal (i.e., three CC tanks) and cesium removal using a single-cycle ion exchange process. The selected technologies provide an initial selection criteria to estimate total curies remaining to be disposed in the vitrified LAW.

Transuranic removal must be provided for three DSTs to meet the Class C concentration limits and is assumed to use a hydroxide precipitation process. Some radionuclide removal will occur during vitrification of the LAW for the volatile species. However, curie removal during melter operation has not been added to the radionuclide removal totals.

Table 5-1. Maximum Practical Key Radionuclide Removal (December 31, 1999, Decay Date).

Radionuclide removal technology	MCi removed
Previous removals	84.1
Solid/liquids separations	52
Single-cycle cesium ion exchange (>0.5 Ci ^{137}Cs/L)	29 ^{137}Cs
Selective hydroxide precipitation for transuranic (3 CC tanks)	0.004 transuranic 0.1 ^{90}Sr

5.2 MEET CLASS C CONCENTRATION LIMITS

The second NRC guideline for classification of the LAW fraction is to incorporate the waste in a solid physical form at a concentration that does not exceed the applicable concentration limits for Class C low-level waste as set out in 10 CFR Part 61. Table 5-2 shows radionuclide concentrations that would result from vitrification of the liquid fraction of tank wastes after the proposed additional separation of radionuclides from the liquid fraction. As shown, radionuclide concentrations in the vitrified waste form would be Class C for transuranics, Class B for ^{137}Cs and ^{90}Sr, and Class A for the others listed.

Table 5-2. *Solidified Waste Radionuclide Concentrations After Supernatant Separations Versus 10 CFR Part 61 Limits.*

Radionuclide	Supernatant inventory after pretreatment[a] (MCi)	Average concentration in low-activity waste glass (Ci/m^3)[b,c]	Class A limit (Ci/m^3)	Class B Limit (Ci/m^3)	Class C Limit (Ci/m^3)
^{137}Cs	5	32	1	44	4,600
^{90}Sr	3.4	22	0.04	150	7,000
Transuranics	0.01	25 nCi/g	10 nCi/g	NLE	100 nCi/g
^{99}Tc	0.032	0.2	0.3	--	3.0
^{79}Se	0.00103	<0.006	NLE	NLE	NLE
^{14}C	0.0053	<0.03	0.8	NLE	8.0
^{129}I	0.000051	<0.0003	0.008	NLE	0.08
^{3}H	0.01	<0.06	40	NLE	NLE
^{126}Sn	0.0016	<0.01	N/A	N/A	N/A
Uranium	0.001	<0.006	NLE	NLE	NLE

NLE = No limit established.

[a]To be conservative, it is assumed that 100 percent of the ^{99}Tc, ^{79}Se, ^{14}C, ^{3}H, ^{129}I, and ^{126}Sn inventories (soluble and insoluble fractions) are incorporated into the immobilized low-activity waste. See text in Section 4.0 for discussion.

[b]The sum of the fractions rule for mixtures of radionuclides has been applied.

[c]The low-activity waste volume is estimated to be 158,000 m^3 of glass.

5.3 MEET DISPOSAL PERFORMANCE OBJECTIVES

The third NRC guideline for classification of the LAW fraction as "incidental" waste is to manage the waste so that safety requirements for disposal performance are comparable to the performance objectives of 10 CFR Part 61. Defense wastes are also managed and disposed under guidance from DOE 5820.2A. The DOE order requires that a PA be prepared for the disposal system. A PA for LAW disposal has not yet been prepared.

The completed PA will identify areas of concern for long-term release of radionuclides to the environment from the disposal system. At that time, mitigating features, such as additional radionuclide removal and/or disposal system barriers, will be evaluated and incorporated into the treatment and disposal system as needed to meet PA requirements.

Although technetium is not a significant contributor to the total activity in the LAW, ^{99}Tc is the largest contributing radioisotope for meeting the maximum 4 mrem/yr criteria, within 10,000 years from closure, from the consumption of groundwater contaminated by the LAW disposal site.

Uranium isotopes and ^{79}Se are minor contributors to the total activity in the LAW but are significant with respect to the disposal performance objectives. ^{79}Se is significant with respect to meeting the maximum 4 mrem/yr criteria, within 10,000 years from closure, from the consumption of groundwater contaminated at the LAW disposal site.

Sensitivity studies of LAW disposal system features indicate that the exposure from ^{99}Tc can exceed the 4 mrem/yr criteria if the LAW glass is disposed without the use of engineered barriers (Mann et al. 1995). The assumptions used in the current environmental assessment calculations are considered conservative. The studies indicate that a LAW ^{99}Tc inventory reduction, of less than a factor of 10, will substantially limit the contribution of the ^{99}Tc to the first dose peak which arrives at about 10,000 years. The LAW inventory reduction can be met by a combination of accurate definition of the tank waste inventory and providing some removal of soluble ^{99}Tc. A combination of engineered barriers, and ^{99}Tc removal can mitigate the potential of exceeding the 4 mrem/yr criteria during the first 10,000 years.

Decisions made for removal of ^{99}Tc, ^{79}Se and uranium isotopes will be based on potential performance assessment impacts, not the total combined activity in the onsite disposed LAW.

5.4 PROPOSED DETERMINATION OF HANFORD SITE TANK WASTE CLASSIFICATION

It is proposed that radionuclides be separated from Hanford Site tank waste so that those remaining in the LAW fraction not exceed the quantities listed in Table 5-3. For comparison purposes, Table 5-3 also includes the previous NRC determination for DST waste only.

Table 5-3. Comparison of Previous and Proposed U.S. Nuclear Regulatory Commission Determinations.

Parameter	Previous NRC determination[a]	Proposed NRC determination[b]
Scope, number of waste tanks	28 DSTs	28 DSTs and 149 SSTs
Low-activity waste (LAW) form	Grout	Glass
Low-activity waste volume, m^3	233,000	158,000
Radionuclides in LAW (MCi)		
^{137}Cs	6 to 7	5
^{90}Sr	1 to 8	3.4
Transuranics	0.002 to 0.01	0.01
^{99}Tc	0.016 to 0.028	<0.03[c]
^{79}Se	--	<0.001
^{14}C	0.0027	<0.0053
^{129}I	0.000033	<.000051
^{3}H	--	<0.01
^{126}Sn	--	<0.0016[c]
Uranium	--	<0.001[c]
Total (without daughters)	7 to 15	8.5

DST = Double-shell tank
NRC = U.S. Nuclear Regulatory Commission
SST = Single-shell tank
-- = No value established
[a]Decay date December 31, 1995
[b]Decay date December 31, 1999.
[c]And as required by the Performance Assessment.

It is concluded that cesium removal from liquid fraction and TRU removal from CC tanks before LAW vitrification, represents radionuclide removal to the maximum extent practical. Therefore, these liquids could be considered "incidental" waste provided the immobilized LAW qualify for disposal in a shallow land disposal facility under DOE waste management requirements comparable to 10 CFR Part 61 requirements. This waste would then be disposed onsite, near surface, as LAW in accordance with the DOE and Washington State Department of Ecology (Ecology) regulations for LLW or Mixed LLW. The residual waste left in the tanks after retrieval is excluded from this determination and will be considered separately at a later time.

6.0 REFERENCES

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10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," *Code of Federal Regulations*, as amended.

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APPENDIX A

CORRESPONDENCE

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APPENDIX A

CORRESPONDENCE

This appendix contains correspondence between the U.S. Department of Energy, the Washington State Department of Ecology, and the U.S. Nuclear Regulatory Commission on the subject of LLW classification. The letters, as described below, are presented in chronological order and document the history of the Hanford Site waste classification issue.

1. November 29, 1988
   Bell, Michael J., Chief Regulatory Branch, Division of Low-Level Waste Management and Decommissioning, Office of Nuclear Materials Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, D.C. to Ronald E. Gerton, Waste Management Division, U.S. Department of Energy, Richland Operations Office, Richland, WA.

   The NRC provides comments on DOE's proposed approach for classifying Double-Shell Slurry Feed (DSSF). In general, the NRC staff support DOE's actions in: (1) seeking an NRC-DOE consensus on the classification of double-shell tank waste; (2) using the source-based concept in classifying waste as HLW or non-HLW; (3) describing the prior treatment of any "incidental" waste; and (4) documenting waste characteristics before classification. Specific comments on the classification approach indicate if DOE could demonstrate that the largest practical amount of the total site activity (expected to be 90 percent or more) attributable to "first-cycle solvent extraction" wastes had been segregated for disposal as HLW, then the NRC would view the residual as a non-HLW. This residual would not be subject to NRC licensing.

2. March 6, 1989
   Rizzo, A.J., Assistant Manager for Operations, Department of Energy, Richland. Washington to R.M. Bernero, Director, Office of Nuclear Materials Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, D.C.

   The Department of Energy, Richland asks the NRC for concurrence that the DST waste planned for disposal by grouting is not HLW, and therefore not subject to NRC licensing. An overall radionuclide material balance for all Hanford Site tank wastes shows that 3 to 5 percent of the key radionuclides which entered the tanks will be disposed of as LLW when the DST wastes are grouted. To meet the NRC criteria of segregating the largest practical amount of activity, DOE also proposes to remove 95 percent of the Cs-137 from the complexant concentrate waste for disposal as HLW. The additional radionuclide removal would reduce the 3 to 5 percent material balance to 2 to 3 percent. The incorporation of ^{137}Cs, ^{90}Sr and TRU into grout would result in radionuclide concentrations comparable to or below Class C (10 CFR 61)

LLW and Class A for ^{14}C and ^{129}I without additional removals. Removal of ^{99}Tc was considered not to be practical or cost effective.

3. September 26, 1989
Bernero, R.M., Director, Office of Nuclear Materials Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, D.C. to A.J. Rizzo, Assistant Manager for Operations, Department of Energy, Richland, Washington.

The NRC agrees that the grout facility for disposal of DST waste would not be subject to NRC licensing. However, the NRC also states that their position on DST waste does not reflect a decision on SST waste, and they will defer judgement on the classification of SST waste until after DOE has completed its waste characterization program.

4. November 17, 1989
Gregoire, C.O., Director, Department of Ecology, State of Washington to K.M. Carr, Chairman, U.S. Nuclear Regulatory Commission, Washington D.C.

The Washington Department of Ecology notifies the NRC that the state of Washington intends to petition the NRC for rulemaking concerning the classification of high-level and "incidental" wastes.

5. January 2, 1990
Husseman, T., Assistant Director, Waste Management, Department of Ecology, State of Washington to S.J. Chilk, Secretary, U.S. Nuclear Regulatory Commission, Washington, D.C.

The state of Washington transmits their petition for rulemaking regarding the classification of HLW at the Hanford Site. The state of Washington asks that the NRC revise the definition of HLW to establish a procedural framework for determining whether certain Hanford Site wastes are HLW or not.

6. December 17, 1990
55 FR 51732, "Definition of the Term 'High-Level Radioactive Waste,'" *Federal Register*.

The above petition for rulemaking regarding the classification of HLW at the Hanford Site is published in the Federal Register.

7. March 2, 1993
Bernero, R.M., Director, Office of Nuclear Materials Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, D.C. to J. Lytle, Deputy Assistant Secretary for Waste Operations, Office of Waste Management, Environmental Restoration and Waste Management, U.S. Department of Energy, Washington, D.C.

The NRC denies the petition for rulemaking submitted by the state of Washington. It is believed the principles for waste classification are well established and can be applied on a case-by-case basis without revision to the regulations. The NRC states that disposal of DST waste would not be subject to NRC licensing based on the assurance that DOE would segregate the largest practical amount of the total site activity for disposal as HLW, leaving behind only a small fraction of moderately radioactive material which would be regarded as "incidental" waste.

8. March 4, 1993
58 FR 12342, Docket No. PRM-60-4, "States of Washington and Oregon: Denial of Petition for Rulemaking." *Federal Register.*

The above petition denial for rulemaking is published in the Federal Register. In the denial text, the NRC refers to information presented in the March 6, 1989 letter from DOE to the NRC (see Item 2 above) and states: "The concentrations of radionuclides in the grout would be comparable to Class C for cesium and transuranic wastes, and to Class A or B for the remainder." The NRC understood this statement to connote that cesium-137 and transuranic radionuclides in the residual waste would be less than the concentration limits for Class C low-level waste, as defined in the NRC's requirements in 10 CFR Part 61, and that the concentration of other radionuclides would be less than the concentration limits for Class A or B low-level waste.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NOV 29 1989

Ronald E. Gerton, Director
Waste Management Division
U.S. Department of Energy
Richland Operations Office
Richland, WA 99352

Dear Mr. Gerton:

We have reviewed the proposed approach for classifying Double Shell Slurry Feed (DSSF) waste, as presented in the September 22nd meeting on the disposal of Hanford defense waste, and offer the following comments for your consideration.

The NRC staff is in general agreement with the basic elements of DOE's proposed approach. In particular, the NRC staff supports DOE's efforts in: 1) seeking a NRC-DOE consensus on the classification of double-shell tank waste; 2) using the source-based concept in classifying waste as HLW or non-HLW; 3) describing the prior treatment of any incidental waste; and, 4) documenting waste characteristics prior to classification.

Regarding the specifics of the waste classification approach described in your presentation, the NRC staff offers the following comments:

(1) The proposed approach states that individual waste streams that contribute less than one percent of tank volume would not be pursued. The NRC staff disagrees with this cutoff since it would not account for potentially significant low volume, high activity waste streams. We think it would be more appropriate to disregard individual waste streams only if they constitute a minimal fraction of tank activity.

(2) DOE would need to provide improved justification for referring to a particular waste as "incidental" waste, and this justification should be consistent with the criteria proposed by NRC in our last meeting (see Enclosure).

(3) The proposed approach indicates that the characteristics of the waste in tank 106 AN would be documented in a report and made available for NRC review. If DOE decides to proceed with such a report, it should be provided for our review sufficiently in advance so agreement can be reached on the classification of waste before it is grouted.

(4) For reasons stated in our September 22nd meeting, the NRC staff does not support the establishment of a NRC/DOE task force to develop a risk based definition for HLW.

As a more fundamental comment on DOE's proposed approach, the staff had hoped that a determination of the classification of DSSF waste in tank 106 AN might apply to the entire DSSF category. We no longer believe that this is practical. Information provided by DOE in our June 9th meeting indicates that individual DSSF tanks contain different suites of wastes. Therefore, the

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documentation and determination of waste classification would need to proceed on a tank-by-tank basis. Such an effort would obviously require a substantial commitment of time and resources, and should be avoided if there is a more efficient way to proceed.

As an alternative approach, we suggest that DOE attempt an overall material balance for HLW at the Hanford site, using the source-based meaning of HLW. It is hoped that this approach might provide a more efficient means of identifying those wastes subject to licensing by NRC under terms of the 1974 Energy Reorganization Act. Under this approach, if DOE could demonstrate that the largest practical amount of the total site activity attributable to "first-cycle solvent extraction" wastes has been segregated for disposal as HLW, then the NRC would view the residual as a non-HLW. We would anticipate that at least 90 percent of the activity would have been separated in this way. Thus, if it can be shown that DOE has processed the waste with the intent to dispose of the HLW in a repository or other appropriate licensed facility, leaving behind only a small fraction of only moderately radioactive material, then the goals stated in 10 CFR Part 50 Appendix F and incorporated in the Energy Reorganization Act would have been satisfied; and the disposal of the residual would accordingly not be subject to NRC licensing.

I hope that these comments are helpful to you in resolving the classification of DSSF and other double-shell tank wastes at Hanford. If you have any questions on this matter or would like to arrange a meeting to discuss the implementation of this alternative, please feel free to contact me on FTS 492-0550.

Sincerely,

J. L. T. Greeniaux for

Michael J. Bell, Chief
Regulatory Branch
Division of Low-Level Waste Management
and Decommissioning
Office of Nuclear Materials Safety
and Safeguards

Enclosure: As stated

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ENCLOSURE

SUGGESTED CRITERIA FOR CLASSIFYING

DECONTAMINATED SALTS AS INCIDENTAL WASTES

(1) SEPARATION PROCESS MUST REPRESENT "GOOD FAITH" EFFORT TO ACHIEVE TOTAL ISOLATION OF HLW FROM NONRADIOACTIVE SALTS.

-- CONCENTRATIONS OF RADIONUCLIDES IN SALTS SHOULD BE COMPARABLE TO CONCENTRATIONS OF OTHER LOW-LEVEL WASTES (NOT JUST BELOW THE CLASS C LIMITS).

-- TOTAL ACTIVITY OF A GROUT DISPOSAL FACILITY SHOULD BE COMPARABLE TO THAT OF OTHER LLW DISPOSAL FACILITIES.

-- AN OVERWHELMING FRACTION (I.E., 5X5) OF THE TOTAL ACTIVITY SHOULD BE CAPTURED IN THE HLW FRACTION, SO THAT THE TOTAL INVENTORY OF ACTIVITY IN GROUT DOES NOT DIFFER GREATLY FROM THE TOTAL INVENTORY OF ACTIVITY AT A TYPICAL LLW DISPOSAL SITE.

(2) AS A PRACTICAL MATTER, THE "GOOD FAITH" EFFORT WILL LIKELY NEED TO BE JUDGED BY CONSIDERING ALTERNATIVE SEPARATION PROCESSES.

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Department of Energy
Richland Operations Office
P.O. Box 550
Richland, Washington 99352

MAR 15 1989

Mr. Robert M. Bernero, Director
Office of Nuclear Materials Safety
and Safeguards
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Bernero:

Members of our staffs have been meeting to discuss the classification and disposal of the Hanford double-shell tank wastes. The central issue is the definition of high-level waste (HLW), and the application of the HLW definition in the Hanford Defense Waste Environmental Impact Statement. Over forty-five years of Hanford operations have resulted in numerous waste streams, involving several treatment activities. This makes classification of wastes somewhat complex. Various approaches have been discussed in detail to resolve the classification issue. The alternate approach suggested in the November 29, 1988, letter from Mr. Michael J. Bell of the Nuclear Regulatory Commission (NRC) to Mr. Ronald E. Gerton of my staff, on classifying the fraction of Hanford's double-shell tank waste planned for disposal by grouting in near-surface vaults, appears to us to be the appropriate way to proceed. This approach utilizes an overall material balance of tank waste at the Hanford site to demonstrate that the largest practical amount of the total site activity attributable to "first-cycle solvent extraction" wastes has been segregated so that only the residuals will be grouted.

An overall radionuclide material balance for all Hanford tank wastes has been prepared and is enclosed. In addition to the HLW, we have included other liquid wastes (low-level waste (LLW) and transuranic waste) which have entered the waste tanks, however, their radionuclide contribution (in curies) is small as compared to the HLW streams. This material balance shows 3-5% of the key radionuclides which entered the tanks will be disposed as LLW in near-surface vaults when the double-shell tank wastes are grouted. The concentration of radionuclides will be comparable to Class C for cesium and transuranic, and to Class A or B for the remainder. The total amount of such waste will be greater than the amount of Class C waste that might be contained in a typical commercial burial ground. It is our understanding that the NRC's concern over the amount of Class C waste is related to the possibility of intrusion and the degree of institutional control to be maintained. Hanford's grout disposal system will provide excellent protection against intrusion since the grout vaults are a solid concrete mass that will have a multilayer engineered barrier on top. In addition, Hanford, as a dedicated government facility, should have a greater likelihood of maintaining institutional controls for relatively long times.

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Based on consideration of alternative separation processes, and to meet the suggested criteria of segregating the largest practical amount of activity, the Department of Energy, Richland Operations Office, proposes to remove additional radionuclides from the double-shell tank waste for disposal as HLW. Specifically, we will remove at least 95% of the Cs-137 from the complexant concentrate waste. This will reduce the 3-5% material balance values to 2-3%.

We believe that the data in the enclosure and discussed in this letter demonstrate that the double-shell tank waste planned for disposal by grouting in near-surface vaults is not HLW, and that NRC licensing is not required. We await your concurrence so that we can proceed with this vital waste disposal program.

If you should have any questions or comments, please feel free to contact me or Mr. Ronald E. Gerton of my staff on (509) 376-1365.

Sincerely,

C. A. Rizzo, Assistant Manager
for Operations

WMD:PKC

Enclosure:
Radionuclide Material Balance

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Enclosure 1

HANFORD TANK WASTE RADIONUCLIDE MATERIAL BALANCE

INTRODUCTION

At the Hanford Site, reprocessing of irradiated nuclear fuel started in 1944. Three uranium and plutonium recovery processes have been used: bismuth phosphate, reduction-oxidation (REDOX), and plutonium-uranium extraction (PUREX).

Each of these facilities generated a source-based high-level waste stream (first-cycle waste from recovery process) and several low-level waste streams including sump wastes, miscellaneous wastes, organic treatment wastes, and decladding wastes.

Supporting operations such as plutonium purification (Plutonium Finishing Plant), equipment decontamination (T Plant), and on-site laboratories generate transuranic (TRU) and low-level waste. Decontamination operations in the Department of Energy's nuclear reactor (N Reactor) also produce low-level waste. These low-level wastes and the treated liquids from the processes described below have routinely been concentrated in evaporators to reduce the waste volume to be stored. The process condensates were discharged to ground (typically after ion exchange to remove entrained Cs-137) while the concentrate is stored for later grout disposal.

The wastes are currently stored in underground single-shell (SST) and double-shell (DST) tanks. The SSTs have been removed from active liquid waste service (some residual liquid is still being removed). The DST waste inventory will be feed to the grout and vitrification disposal processes.

WASTE PARTITIONING

First-cycle waste generated prior to 1983 has gone through a combination of treatments such that the supernate of the treated waste is considered low-level waste. The treatments include neutralization/decantation, cesium removal, and strontium removal (see Figure 1).

Neutralization/Decantation

All facility wastes stored in underground tanks have always been neutralized with sodium hydroxide to a caustic endpoint. The neutralization precipitates the actinides and the bulk of the fission products including strontium-90 (Sr-90). The solids settle in the 75-foot diameter tanks and form a sludge layer. Important radionuclides that are primarily soluble and remain in the alkaline supernate are carbon-14 (C-14), technetium-99 (Tc-99), iodine-129 (I-129), and cesium-137 (Cs-137).

The settled solids remain in the tanks when the supernate is decanted and transferred for further processing. Special efforts such as sluicing or agitation with mixing pumps are required to transfer the settled solids.

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Cesium Removal

Cs-137 has been removed from HLW supernate via several processes. The first-cycle wastes from the bismuth phosphate process were treated with nickel ferrocyanide in the mid-1950s to precipitate the cesium in some single-shell tanks.

The alkaline supernates generated by the REDOX and PUREX plants prior to 1968 were processed in B Plant for cesium removal by ion exchange. This process will also be used to remove cesium from PUREX Plant alkaline supernates generated since 1983. During the period from 1968 until the PUREX Plant shutdown in late 1971, the acidic PUREX Plant HLW was routed to B Plant without neutralization. A phosphotungstic acid (PTA) precipitation process was then used in B Plant to remove the cesium.

The cesium recovered in B Plant was purified and doubly encapsulated for long-term storage. This cesium will be disposed in a geologic repository.

Strontium Removal

As discussed previously, the routine neutralization/decantation of first-cycle extraction waste provides a Sr-90 and actinide separation. However, some first-cycle extraction waste had further strontium removal performed. The Sr-90 in acid PUREX Plant HLW was removed in B Plant via a solvent extraction process concurrently with the cesium PTA precipitation. The B Plant residual wastes were then neutralized and routed to the underground storage tanks.

The settled sludge produced from the neutralized PUREX Plant first-cycle extraction waste prior to 1968 was sluiced, acidified, and processed for Sr-90 removal via the B Plant solvent extraction process.

The strontium recovered in B Plant was doubly encapsulated for long-term storage. This strontium will also be disposed in a geologic repository.

The PUREX Plant HLW sludge (containing the Sr-90 and transuranic nuclides) generated since the 1983 restart will be separated from the supernate. The sludge, along with Cs-137 removed from the supernate, will be incorporated into glass in the Hanford Waste Vitrification Plant (HWVP) for disposal in a geologic repository.

Transuranic Removal

The organic complexants used in the B Plant solvent extraction process to recover strontium tend to solubilize the strontium and actinides and inhibit their precipitation. The complexant concentrate (CC) waste contains the bulk of the waste from B Plant strontium recovery and will be treated in B Plant with the transuranic-extraction (TRUEX) process to remove transuranics (TRU) which will be routed to HWVP for disposal in a geologic repository.

The sludge from the PUREX Plant neutralized cladding removal waste (NCRW) and the solids from the Plutonium Finishing Plant (PFP) wastes contain greater than 100 nCi/g TRU nuclides. These sludges will also be treated in B Plant with the TRUEX process to remove TRU which will be routed to HWVP for disposal in a geologic repository.

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HANFORD WASTE PARTITIONING

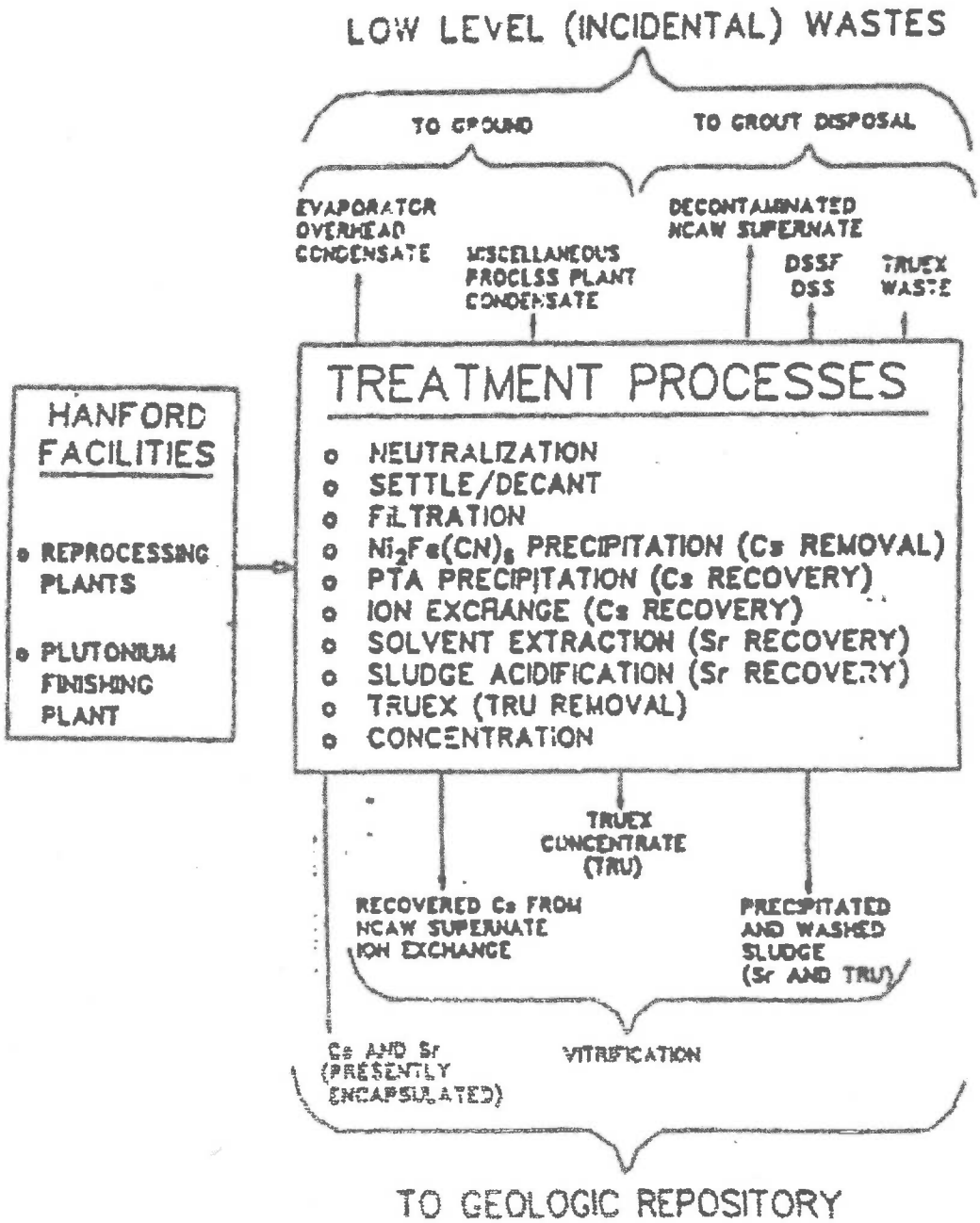

FIGURE 1

RADIONUCLIDE MATERIAL BALANCE

The estimated percentage disposition of the Sr-90, Cs-137, TRU nuclides and total activity from Hanford reprocessing wastes is shown in Figure 2.

The bases for the radionuclide inventories used to calculate the percentages are estimates of the radionuclide inventory that entered the SST and DST wastes prior to 1983, the measured inventory in strontium and cesium capsules, and estimates of the radionuclides in wastes produced after the PUREX Plant restarted in 1983.

The pre-1983 SST/DST estimates are from the Hanford Defense Waste Environmental Impact Statement (DOE/EIS-0113) and the Hanford input to the Integrated Data Base (IDB) Document for 1989 (DOE/RW-0006, Rev. 4). The IDB document was the source for the strontium/cesium capsule inventory. Both documents used the Track Radioactive Components (TRAC) System as the primary source for the radionuclide inventory in the Hanford tanked waste. Input data for the TRAC System were historical records of reactor, reprocessing, and waste management operations. The computer programs RISO-2 and DCCDE were used to calculate the radionuclides in the irradiated reactor fuel. The TRAC model then used these inputs to calculate the waste generated by the reprocessing plants as a function of time and follow the waste through the various waste management processes and transactions to arrive at the final waste tank inventories.

The radionuclide inventories for post-1983 wastes were estimated using the ORIGEN-2 computer program to calculate the radionuclides in all irradiated N Reactor fuel to be reprocessed following the 1983 restart of the PUREX Plant. The PUREX Plant, waste management and planned B Plant pretreatment flowsheets were used to estimate the partitioning of the radionuclides between HLW (HWVP feed) and LLW (grout feed).

The radionuclide inventories as shown in the EIS and IDB documents are best estimates but are subject to uncertainties because of factors such as:

- incomplete/inaccurate records dating back to 1944;
- the use of bounding fuel impurity limits to calculate maximum activation products, i.e. C-14, since actual analyses are no longer available; and
- incomplete understanding of the chemistry and resulting pathways of some elements in the reprocessing and waste treatment processes.

Ideally, actual analytical data for all Hanford tanked waste would be available. All grout and vitrification feeds will have such analyses before they are processed. However, only limited analytical results are currently available. These analyses indicate that the total Cs-137 that may exist in grout feed could be as much as 20 million curies (Ci) vice the 12-13 million Ci which is the basis for the 5-6% range shown in Figure 2. (Both of these inventories are corrected for decay to the end of CY 1995.)

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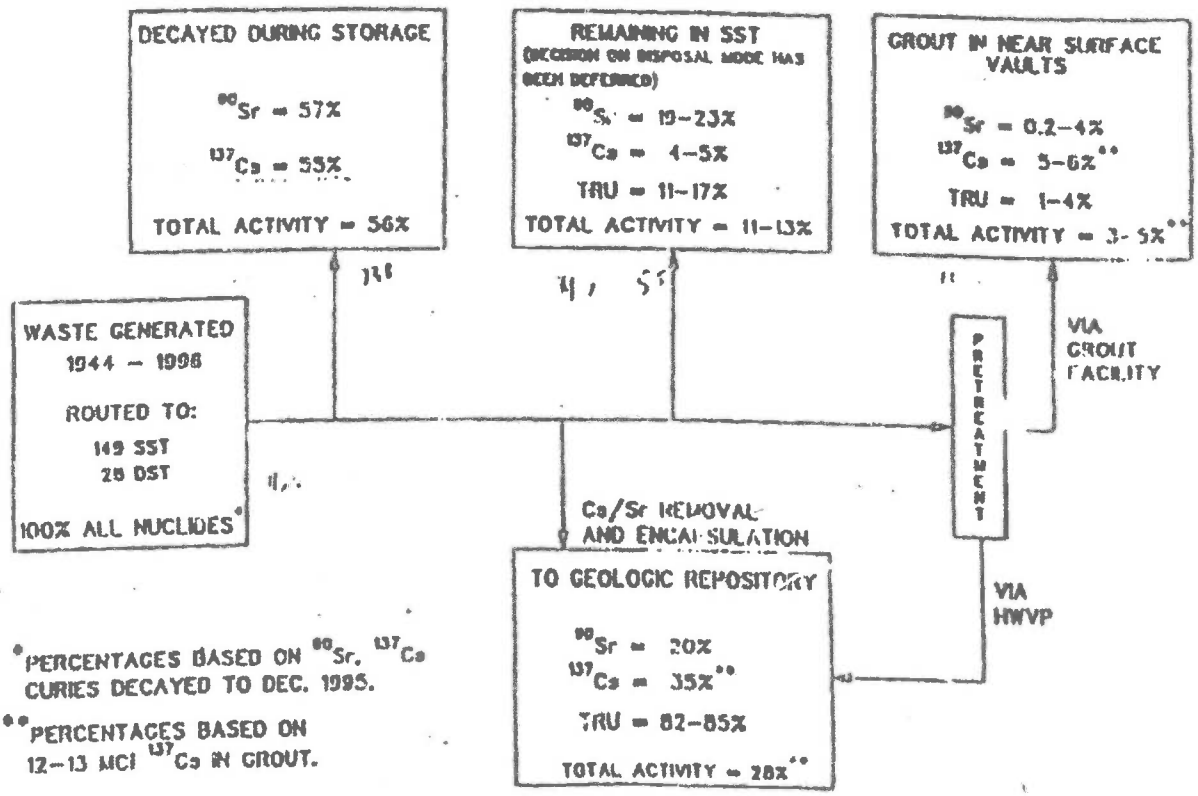
FIGURE 2

RADIONUCLIDE MATERIAL BALANCE
HANFORD TANK WASTE



*PERCENTAGES BASED ON ^{90}Sr, ^{137}Cs CURIES DECAYED TO DEC. 1995.

**PERCENTAGES BASED ON 12–13 MCI ^{137}Cs IN GROUT.

PLANNED PRETREATMENT OF GROUT FEEDS

Pretreatment in B Plant is planned for five of the three Hanford waste types before they are considered to be suitable grout feeds. These waste types are the neutralized current acid waste (NCAW-PUREX Plant first-cycle extraction waste), the neutralized cladding removal waste (NCRW) solids, the Plutonium Finishing Plant (PFP) waste solids and complexant concentrate (CC).

The NCAW will be transferred to B Plant where the solids containing the Sr-90 and actinides will be removed and the clarified supernate ion exchanged for Cs-137 removal. The washed NCAW solids and the recovered Cs-137 will be routed to HWVP and the decontaminated supernate stored for grout feed.

The NCRW and PFP solids are TRU wastes. These solids will be transferred to B Plant, dissolved in acid and then treated using the TRUEX extraction process. The TRU concentrate from the TRUEX process and any undissolved solids will be routed to HWVP.

Some stored complexant concentrate is TRU waste. Present plans are to treat all the CC in B Plant for complexant destruction and then TRU removal by the TRUEX process. The TRU concentrate and any undissolved solids will be routed to HWVP. These plans may be modified if further characterization determines some of the wastes are non-TRU or that complexant destruction is not required.

The non-TRU wastes from the above pretreatment processes are considered to be suitable LLW grout feeds. Waste solids would be routed directly to a grout feed tank while the supernates may be further concentrated.

No additional treatment is planned for double-shell slurry feeds (DSSF). The DSSF is produced by evaporation of the dilute Hanford LLW; e.g. NCRW supernate, PFP supernate, PUREX and B Plant miscellaneous wastes, pretreatment wastes and SST residual liquids.

Additional treatment of double-shell slurry (DSS) is not planned. The DSS is produced by further concentration of DSSF past the sodium aluminate phase boundary. This stored waste contains solids and dilution/dissolution will be required for retrieval.

POTENTIAL ADDITIONAL RADIONUCLIDE REMOVAL

Estimated curies disposed to grout for key radionuclides are shown in Table 1. The incorporation of these nuclides into grout would result in radionuclide concentrations comparable to or below Class C (10 CFR 61) LLW.

The estimated concentrations of C-14 and I-129 in the grout would be comparable to Class A (10 CFR 61) limits. There is no viable technology available to further remove these nuclides from the large volume of grout feed. As previously indicated, the inventory of C-14 is probably overstated. The estimated maximum concentration of Tc-99 in grout feed (in the pretreated NCAW supernate) would be near or slightly above the Class A limit. Additional removal of Tc-99 via an ion exchange process is theoretically possible. However, further removal does not appear practical or cost effective. Accordingly, no pretreatment for removal of these nuclides is planned.

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Transuranic wastes will be pretreated for TRU removal. The treated grout feed would be comparable to NRC Class C LLW.

The inventories of Sr-90 and Cs-137 dominate the estimated total activity to be disposed to grout. The inventories of these nuclides estimated for grout disposal are not significant in the Performance Assessment for grout due to their relatively short half-lives (Sr-90 = 28.5 years, Cs-137 = 30.0 years). The estimated maximum concentration of Sr-90 plus Cs-137 in grout would be a factor of approximately 30 below the Class C LLW limit. However, possible additional removal processes were examined to assess the practicality of further removal of these nuclides from grout feed.

The Sr-90 inventory estimates are uncertain, but the bulk of this nuclide is in the complexant concentrate (CC) which contains a large fraction of precipitated solids. The CC is treated with the TRUEX process to reduce the quantity of solids routed to HWVP for incorporation into glass. After acidification of the CC in B Plant to dissolve solids, complexant destruction, and TRU removal by the TRUEX process, the Sr-90 would remain in solution. It then would be routed to grout with the TRUEX waste. If it is necessary to remove the Sr-90, the TRUEX process would not be used but rather the Sr-90 and TRU solids would be routed to HWVP by neutralization/precipitation after complexant destruction. However, the total volume of vitrified waste would be increased (approximately 20%) and additional costs of about $150 million would be incurred. It is likely that alternative processes could be developed to reduce these additional volumes and costs.

Smaller quantities of Sr-90 are contained in the DSSF. The presence of low concentrations of complexant in the DSSF precludes further Sr-90 removal by precipitation processes. Destruction of the dilute complexant in DSSF in B Plant for Sr-90 removal would delay the grout program by about 5 years while process equipment was designed, fabricated and installed and the initial grout feed batch was processed. The complexant destruction would require approximately four years of B Plant operation and increase B Plant and HWVP operating costs 100 to 200 million dollars.

The cesium nickel ferrocyanide precipitation process used previously in Hanford SSTs could effectively remove Cs-137 from tanked waste. However, the ferrocyanide precipitate, if routed to HWVP, could result in potential explosion hazards and metal phase production in the glass melter which would not be tolerated.

The presence of potassium in some Hanford DSSF precludes the use of the SRP process of cesium tetraphenyl boron precipitation since the potassium would be co-precipitated with the cesium. At least two orders of magnitude more tetraphenyl boron (i.e., thousands of tons) would be required for treatment of the Hanford waste as compared to the SRP waste, with the attendant problems involved in handling this benzene compound.

Ion exchange remains as the only technically viable process for removing the Cs-137. Use of this process is already planned for the HCAW supernate, with startup scheduled for 1992 in B Plant. The process could be used on other grout feeds, but there are timing and logistics problems. If ion exchange removal of Cs-137 were fully utilized for all grout feeds, the volume of ion exchange feed would increase from about seven million gallons to approximately

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- 3 -

TABLE 1

ESTIMATED RADIONUCLIDES DISPOSED TO GROUT

Nuclide	MCi
C-14	0.0027
Tc-99	0.016 - 0.025
I-129	33×10^{-6}
Sr-90*	1 - 8
Cs-137*	12 - 13
TRU	0.002 - 0.01
Total Activity**	13 - 21

*The Sr-90 and Cs-137 curies are decayed to the end of CY 1995

**Total activity taken as sum of Sr-90 plus Cs-137 since these two nuclides (and their daughter products) dominate the total inventory.

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35 million gallons. This increased volume represents approximately 15 years additional operation with the present ion exchange system design or 10 years additional operation with an upgraded system. The grout volume would be increased about 10%.

If the tanks that are now available for grout feed are held for ion exchange processing (which cannot start before 1992), the grout program would have to shut down. These tanks then would not be emptied and made available to store waste currently being generated. A waste storage space shortfall would result. New waste storage tanks could not be provided in time to support Hanford programs, including removal of residual liquids from SSTs and production operations.

Full ion exchange treatment of all grout feeds would route approximately 12.8 MCi of Cs-137 to glass rather than grout. This would reduce the total radioactivity disposed to grout by about 51%. However, Cs-137 is not the radionuclide of primary concern for the grout long-term performance assessment, and the Cs-137 removal would not significantly enhance the safety of the grout disposal system.

The cost for Cs-137 removal from HCAW supernate in B Plant will be approximately $3/Ci (allocated operating cost). The estimated RCM costs (incremental capital plus allocated operating) for Cs-137, Sr-90 removal from the remaining grout feeds are as follows:

Cs-137 Removal	MCi Removed	$/Ci
DSSF/DSS	6-7	25-30
CC	~6	~8
Sr-90 Removal		
DSSF/DSS	0.25	~900
CC	1-8	20-150

The above costs would increase if detailed analysis of facility operating schedules (e.g., PUREX, B Plant, HWVP) determined that unproductive standby time for a facility was unavoidable.

CONCLUSION

As discussed previously, delaying the grout processing of currently available feeds, i.e., DSSF/DSS, will severely impact Hanford site programs. However, pretreatment of the CC in B Plant is not scheduled to begin until CY 2000. Thus, time is available for flowsheet development and design/installation of additional equipment as necessary to remove Cs-137 from the CC. The removal and vitrification of the Cs-137 from CC is expected to result in less than 10 additional canisters for repository disposal, a minimal impact. Figure 3 shows that if the approximately 6 MCi Cs-137 contained in the CC is removed, the percentage of Cs-137 and total activity in the grout would drop to approximately 3% and 2-3% respectively. This additional pretreatment is now being proposed.

It is concluded that with the addition of Cs-137 removal from CC, the planned grout feed pretreatment represents a "good faith" effort to maximize isolation of HLW from Hanford site LLW.

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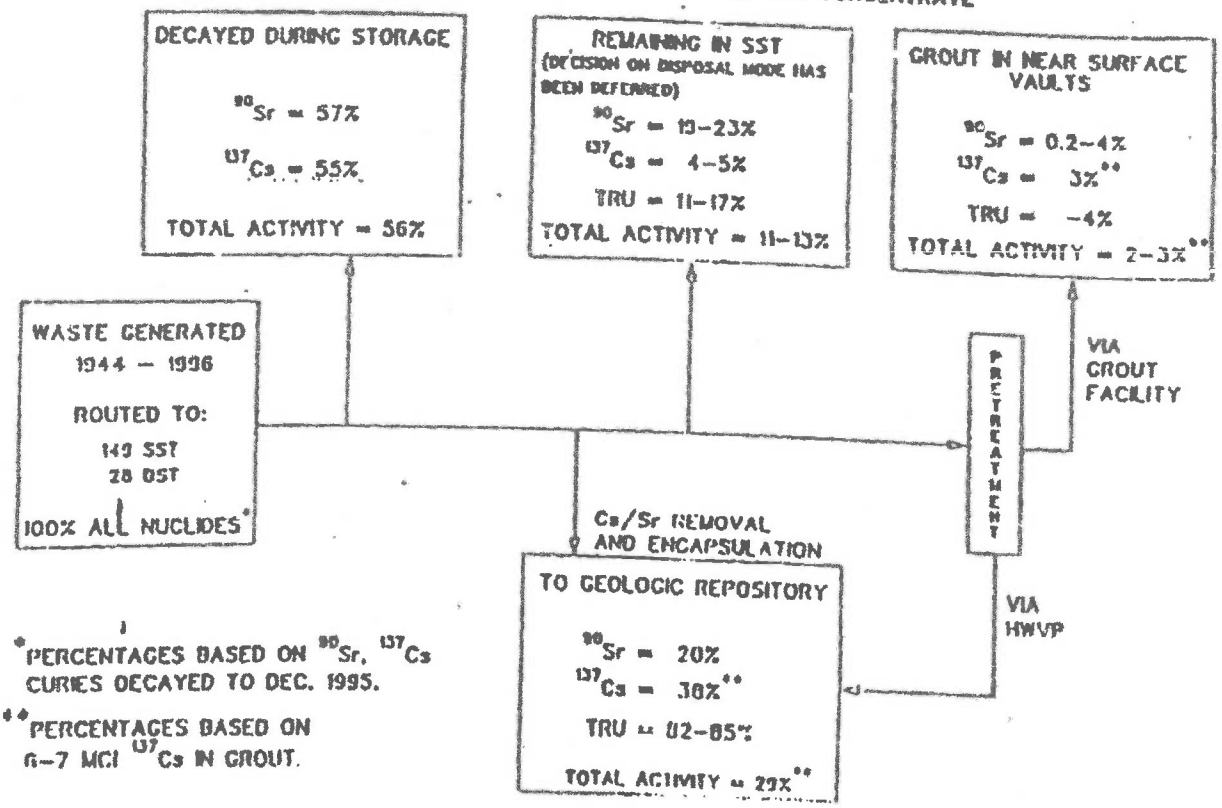
FIGURE 3

RADIONUCLIDE MATERIAL BALANCE HANFORD TANK WASTE

IF ADDITIONAL ^{137}Cs REMOVED FROM COMPLEXANT CONCENTRATE



*PERCENTAGES BASED ON ^{90}Sr, ^{137}Cs CURIES DECAYED TO DEC. 1995.

**PERCENTAGES BASED ON 6–7 MCI ^{137}Cs IN GROUT.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SEP 1 1 1989

Mr. A. J. Rizzo
Assistant Manager for Operations
U.S. Department of Energy
Richland Operations Office
P.O. Box 550
Richland, Washington 99352

Dear Mr. Rizzo:

We have reviewed your letter dated March 6, 1989 concerning the classification and disposal of the Hanford double-shell tank waste. Your letter and supporting information assert that the double-shell tank waste planned for disposal by grouting in near-surface vaults is not high-level waste (HLW), and that U.S. Nuclear Regulatory Commission (NRC) licensing is not required. Your letter requests NRC concurrence in this position.

As you know, our staffs have met on several occasions over the past year in an effort to determine which of the Hanford tank wastes are properly classified as HLW. We consider that the applicable definition of HLW, for purposes of classifying the Hanford tank wastes, is that set forth in 10 CFR Part 50, Appendix F. Specifically, HLW is defined as "those aqueous wastes resulting from the operation of the first cycle solvent extraction system or equivalent, and the concentrated waste from subsequent extraction cycles, or equivalent, in a facility for reprocessing irradiated reactor fuels."

The rulemaking record for Appendix F specifically recognizes a number of "incidental," non-HLW waste streams associated with reprocessing plant operations. These include cladding hulls, ion exchange media, sludges, and miscellaneous trash generated during reprocessing operations. Not mentioned, however, are wastes resulting from further processing of HLW (e.g., volume reduction) or removing non-radioactive materials that were added to the HLW for improved processing and/or storage (e.g., the addition of alkaline material to neutralize acidic HLW). At West Valley and the Savannah River Plant, NRC has agreed that such wastes are not HLW. At Hanford, the question of waste classification (and NRC licensing authority) has been complicated by the mixing of waste from various sources over the past 45 years. This mixing has changed the original characteristics of the wastes and has resulted, in some cases, in the mixing of HLW and low-level waste (LLW). Consequently, it is now difficult to directly differentiate between HLW and LLW, using the source-based definition of Appendix F.

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In earlier meetings of our staffs, criteria were suggested for determining when such wastes should be classified as "incidental" wastes rather than as HLW, and these criteria were documented in our letter of November 29, 1988. Your March 6, 1989 letter records U.S. Department of Energy's (DOE's) application of these criteria. Specifically, your letter proposes that the bulk of the key radionuclides (i.e., strontium, cesium and transuranics) would be separated for disposal in a geologic repository, so that only three to five percent of the original inventories of these radionuclides would be disposed by grouting in near-surface vaults. Your letter also states that the concentration of radionuclides in the grout will be comparable to Class C LLW as defined by 10 CFR Part 61 for cesium and transuranics, and to Class A or B for the remainder. Finally, your letter evaluates the practicability and cost-effectiveness of additional radionuclide removal. An additional separation process, beyond those originally contemplated, was found to be cost-effective for removal of an additional six million curies of cesium. It is that would further reduce the total activity disposed in the grout facility to two to three percent of the inventory of HLW that originally entered the tanks. DOE is now proposing to perform this additional radionuclide removal to improve the isolation of HLW. The NRC agrees that the criteria used by DOE for classification of the grout feed as LLW are appropriate. Therefore, the grout facility for the disposal of the double-shell tank waste would not be subject to our licensing authority.

Your letter indicates that the radionuclide inventory is an estimate based on existing computer models, rather than actual analyses of tank waste. Given the uncertainty in the actual radionuclide inventory, we endorse your plans to sample and analyze the grout feeds before disposal in an effort to control the final composition of the grout feed. If, in the course of conducting this sampling program, you find that the inventories of key radionuclides entering the grout facility are significantly higher than you now estimate, you should notify us so that the classification of the waste can be reconsidered. The NRC requests that DOE periodically submit summaries of the analytical results of all the samples to NRC and other affected parties in a timely manner.

Our position on the double-shell tank waste should not be interpreted to reflect a decision on disposal of single-shell tank waste or to establish a precedent in any other context. We intend to defer judgment on the classification of single-shell tank waste until after DOE has completed its program of characterizing this waste. We anticipate that final documentation will be issued for public comment before a decision is made on the disposal of single-shell tank waste.

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Mr. A. J. Ritzo - 3 -

If you should have any questions or comments about this letter, please contact me or Dr. Michael J. Bell, Chief, Regulatory Branch, of my staff at (301) 492-0520.

Sincerely,

Robert M. Bernero, Director
Office of Nuclear Material Safety
and Safeguards

cc: Terry Husseman
WA Department of Ecology
William Don Tahkeal
Yakima Indian Nation
Jeff Brockel
Oregon/Washington Liaison

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CHRISTINE O. GREGOIRE
Governor

STATE OF WASHINGTON
DEPARTMENT OF ECOLOGY
Mail to PV-11 • Olympia, Washington 98504-8711 • (206) 459-6000

November 17, 1989

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

The purpose of this letter is to notify you that the state of Washington intends to petition the U.S. Nuclear Regulatory Commission (USNRC) on a matter of great importance. Specifically, this petition for rulemaking (under subpart H of 10 CFR Part 2) will address the USNRC's approach to the classification of high-level and "incidental" wastes. Through it, we intend to stress the need for a credible definition of "incidental" waste, one which is based on a thorough and sound technical evaluation, and which is subjected to independent peer review and close public scrutiny.

This action is being prompted by a number of discussions and correspondence between our respective staff. Most recently, on August 4, 1989, Washington Department of Ecology staff met with USNRC staff to review a March 6, 1989 U.S. Department of Energy (USDOE) proposal concerning the classification of high-level wastes as it relates to USDOE's Hanford double-shell tanks. Following this meeting, Terry Husseman of my staff summarized our major conclusions, comments, and concerns in an August 15, 1989 letter to Robert M. Bernero (USNRC).

Mr. Bernero's response was contained within letters dated September 25, 1989 to Mr. Husseman and to Mr. A.J. Rizzo of the USDOE. In summary, Mr Bernero declined any USNRC oversight role at Hanford, and changed the classification of some Hanford double-shell wastes from high-level to low-level waste (letters enclosed). Our request that the USNRC define its actions through publication within the federal register, and through appropriate opportunity for public comment was also denied. It is our feeling that the USNRC, as the agency responsible for the licensing of high-level radioactive waste disposal, should reconsider this action.

I want to emphasize that USDOE's noted approach to the management of its tank wastes (March 6, 1989) is generally consistent with the terms of the Hanford Federal Facility Agreement and Consent Order. As such, moving ahead with the grout disposal program is essential to Hanford cleanup, and care should be taken in order that this program not be delayed by legal challenges or because of a lack of public participation.

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We believe that issues relating to: the classification and management of these wastes, the appropriate level of USNRC involvement, and associated public participation, are too important to be implemented by letter from Commission staff.

I have consequently asked my staff to prepare appropriate petition documentation, and hope to submit it to you no later than January 1, 1990.

If you have questions or comments regarding this issue, please contact me or Mr. Terry Husseman at (206) 459-6168 and 459-6029 respectively.

Sincerely,

Christine O. Gregoire

Christine O. Gregoire
Director

Enclosures

cc: Samuel J. Chilk
Dan Silver
Mike Lawrence
Robie Russell

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CHRISTINE O. GREGOIRE
Director

STATE OF WASHINGTON

DEPARTMENT OF ECOLOGY

Mail Stop PV-11 • Olympia, Washington 98504-8711 • (206) 459-6000

January 2, 1990

Mr. Samuel J. Chilk, Secretary
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ATTENTION: Chief, Docketing and Service Branch

Dear Mr. Chilk:

The purpose of this letter is to transmit to you our petition for rulemaking (under subpart H of 10 CFR Part 2) regarding the classification of high-level radioactive wastes at the U.S. Department of Energy's Hanford site. Please also refer to my November 17, 1989 notice of intent which was addressed to Chairman Kenneth M. Carr of the Commission.

Note that the enclosed petition represents the combined views of both Washington and Oregon, and of the Yakima Indian Nation. We each look forward to working with you and Commission staff on this very important issue.

Sincerely,

Terry Husseman
Assistant Director
Waste Management

THhji

cc: Oregon and Washington Congressional Delegation
Dan Silver
Michael Lawrence
David Yaden
Cecil Sanchey

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
BEFORE THE COMMISSIONERS

PETITION FOR RULEMAKING
AND
REQUEST FOR INSTITUTION OF A RULEMAKING PROCEEDING

The states of Washington and Oregon, and the Yakima Indian Nation ("Petitioners") hereby respectfully request and petition the Nuclear Regulatory Commission ("the Commission") to exercise its rulemaking authority, offer adequate opportunity for public comment, and adopt a regulation concerning classification of Hanford high-level radioactive wastes currently stored in retrievable, surface, storage facilities. Procedures governing the rulemaking process are found at 5 U.S.C. Section 533 and 10 C.F.R. Section 2.800-2.809.

I. Proposed Rules

The petitioners ask that the Commission amend 10 C.F.R. Part 50 to add language clarifying that all Hanford double-shell tank wastes are high-level radioactive waste, unless the Commission on a tank by tank basis determines the following:

1. That the U.S. Department of Energy (USDOE) has demonstrated that the largest technically achievable amount of activity from each tank has been isolated for vitrification prior to permanent disposal.

2. That the heat produced by residual radionuclides, together with the heat of reaction during grout processing, will be within limits established to ensure that grout meets temperature requirements for long term stability for low-level waste forms.

3. That any other pretreatment processes (e.g. TRUEX) have undergone appropriate evaluation by the Commission prior to implementation.

II. Grounds and Interest

This rulemaking petition is based, in large part, on Section 202 of the 1974 Energy Reorganization Act, which defines Commission authority over retrievable surface storage facilities and other facilities authorized for the express purpose of subsequent long-term storage of high-level radioactive waste generated by the Administration, (now Department of Energy) which are not used for, or are part of, research and development activities. The grounds and interests of the state are based on the following facts and issues:

1. The USDOE publication titled Integrated DataBase for 1988: Spent Fuel and Radioactive Waste Inventories, Projections, and Characteristics states that high-level waste (HLW) which is generated by the reprocessing of spent reactor fuels and irradiated targets, generally contains more than 99 percent of the nonvolatile fission products produced in fuel or targets during reactor operation. The HLW from a facility that recovers uranium and plutonium contains approximately 0.5 percent of these elements. The inventories of HLW that is in storage in the 28 Hanford double-shell tanks at the end

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of 1987, contained 116,000,000 curies in 19,400,000 gallons (73,400 cubic meters) of waste. Thus, it is clear that even USDOE now recognizes that the double-shell tank waste is HLW.

2. HLW was first defined in 1970 by the Atomic Energy Commission (AEC) in terms of the source of the material rather than by its hazardous characteristics. The AEC defined HLW as:

"those aqueous wastes resulting from the operation of the first cycle solvent extraction system, or equivalent, and the concentrated wastes from subsequent extraction cycles, or equivalent in a facility for reprocessing irradiated reactor fuels."

35 Fed. Reg. 17530, 17532, Nov. 14, 1970. 10 CFR Part 50, Appendix F. This definition of HLW has not been changed and both USDOE and the Commission currently utilize this definition.

3. Over the last 45 years, mixing of wastes from different sources has complicated the classification of Hanford tank wastes, including double-shell tank wastes. Moreover radionuclide inventories are estimates, and subject to substantial uncertainty. Variables include incomplete and inaccurate records, the lack of actual fuel and/or waste analyses, and incomplete understanding of the chemistry and pathways in reprocessing and waste treatment processes. Thus, neither USDOE, the Commission or the petitioners have adequate information regarding the radioactive portion of the double-shell tank waste.

4. USDOE plans to pretreat Hanford double-shell tank waste in order to partition wastes into a "high-level" portion for vitrification and a low-level or "incidental" portion which will subsequently be disposed of at the Hanford Grout facility (currently under construction). USDOE estimates that from a minimum of 13,000,000 curies to more than 21,000,000 curies will be disposed of at the Grout facility.

5. The petitioners have a strong vested interest in the safe, permanent disposal of Hanford high-level tank wastes. There is great uncertainty at this time, concerning the ability of USDOE to demonstrate that the largest technically achievable amount of activity from each tank can be or will be isolated for vitrification. This is evidenced by USDOE's exceptionally large uncertainty (between 13,000,000 and 21,000,000 curies) concerning residual activity scheduled for surface disposal via grout. (Table 1 from Enclosure 1 of a March 6, 1989 letter from Mr. A. J. Rizzo to Mr. Robert M. Bernero.)

6. The petitioners believe the Commission has the legal authority and obligation to promulgate regulations concerning the classification of Hanford tank wastes and that Commission staff are uniquely qualified to evaluate the appropriateness and effectiveness of various pretreatment processes. The rule amendment suggested above would authorize the Commission to ensure that Hanford double-shell tank wastes be subject to safe and appropriate methods of pretreatment.

7. The petitioners have an interest in and a prevailing responsibility for the protection of the future health and safety of the citizens of the Pacific Northwest. The rule amendment suggested above is essential to provide this protection.

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III. Statement in Support

HLW was first defined in 1970 by the Atomic Energy Commission (AEC) in terms of the source of the material rather than by its hazardous characteristics. The AEC defined HLW as:

> "those aqueous wastes resulting from the operation of the first cycle solvent extraction system, or equivalent, and the concentrated wastes from subsequent extraction cycles, or equivalent in a facility for reprocessing irradiated reactor fuels."
>
> 35 Fed. Reg. 17530, 17532, Nov. 14, 1970. 10 CFR Part 50, Appendix F.

The term HLW was first used by Congress in the Marine Protection Research and Sanctuaries Act of 1972, P.L. 92-532, as amended by P.L. 93-254 (1974) codified at 33 U.S.C. Section 202 of ERA. See 52 Fed. Reg. 5992, 5993.

The term HLW is not defined in the Energy Reorganization Act (ERA). The starting point in defining HLW for ERA purposes is found in existing regulations adopted by the Atomic Energy Commission (AEC) in 1970. In this regard, Congress can be presumed to have been aware of the existing regulatory definition when it used the term HLW in Section 202 of ERA. The existing definition adopted by the AEC in 1970 (10 CFR Section 50, Appendix F) as noted earlier, focuses on the process by which HLW is produced to define HLW.

It is clear that defense wastes at Hanford, including double-shell tank wastes, were considered HLW when the ERA was passed. The legislative history of the ERA reveals Congress' intent at that time to subject defense facilities to NRC oversight. In fact, the Congress recognized that HLW was leaking from temporary AEC tanks at various facilities and intended to subject such wastes to NRC licensing at the point of application of a permanent waste management solution. Moreover, the House Committee on Energy and Commerce found that "existing law with respect to atomic energy defense activities is unchanged by this Act, and facilities for the disposal of waste from defense activities remain subject to licensing by the Nuclear Regulatory Commission." H. Rep. No. 785, pt. 1, 97th Cong., 2d Sess., 38, August 20, 1982.

This issue was considered in USDOE's Final EIS on defense wastes at Hanford. The Environmental Impact Statement considered various alternative methods for dealing with these defense wastes. USDOE recognized that:

> "Further, Section 202 of the Energy Reorganization Act requires Commission licensing of those DOE facilities authorized for the express purpose of long-term storage of high-level radioactive waste which are not used for, or are not a part of, research and development activities. Therefore, to the extent that any decision based on this final EIS requires defense high-level waste to be placed in a repository constructed under the Nuclear Waste Policy Act, or a facility subject to licensing under Section 202 of the Energy Reorganization Act such a repository or facility would be subject to licensing by the Commission."
>
> Final Environmental Impact Statement, Disposal of Hanford Defense High-Level, Transuranic and Tank Wastes, vol. 1, 6.11, December 1987.

In short, under existing law, Hanford double-shell tank waste is HLW. A letter from the Commission cannot change the law. Consequently, long term storage or disposal of double-shell tank waste is currently

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subject to licensing by the Commission. However, the rule amendment suggested above would authorize and require appropriate Commission oversight of the management of Hanford double-shell tank wastes, but would allow, under certain circumstances the avoidance of the admittedly cumbersome licensing process.

Petitioners believe that the rule amendment suggested would protect human health and the environment, and would facilitate meaningful Commission involvement in the ultimate disposal and/or long term storage of Hanford double-shell tank waste.

PETITIONERS:

For the State of Washington

Terry Husseman
Assistant Director, Waste Management
Washington Department of Ecology

For the State of Oregon

David Yaden
Director
Oregon Department of Energy

For the Yakima Indian Nation

Cecil Sanchey
Chairman, Radioactive
Hazardous Waste Committee
Yakima Indian Nation

Dated: JAN 0 2 1990

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Definition of Term "High-Level Radioactive Waste"

NRC Docket No. PRM-60-4

55 FR 51732

for at least 3 years, unless a longer retention time is required by part 75 of this chapter.

(2) Records that must be maintained pursuant to this part may be the original or a reproduced copy or a microform if such reproduced copy or microform is duly authenticated by authorized personnel and the microform is capable of producing a clear and legible copy after storage for the period specified by Commission regulations. The record may also be stored in electronic media with the capability for producing, on demand, legible, accurate, and complete records during the required retention period. Records such as letters, drawings, and specifications must include all pertinent information such as stamps, initials, and signatures.

(3) The licensee shall maintain adequate safeguards against tampering with and loss of records.

Dated at Rockville, Maryland, this 11th day of December 1990.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,

Secretary of the Commission.

[FR Doc. 90-[illegible]7 Filed 12-14-90; 8:45 am]

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10 CFR Part 50

[Docket No. PRM-50-4]

Definition of the Term "High-Level Radioactive Waste"

AGENCY: Nuclear Regulatory Commission.

ACTION: Petition for rulemaking.

SUMMARY: The States of Washington and Oregon request that the Commission revise the definition of the term "high-level radioactive waste" so as to establish a procedural framework and substantive standards by which the Commission will determine whether reprocessing waste, including in particular certain waste stored at the U.S. Department of Energy's site at Hanford, Washington, is high-level radioactive waste and therefore subject to the Commission's licensing authority.

DATES: Submit comments by March 18, 1991. Comments received after this date will be considered if it is practical to do so, but consideration cannot be given except as to comments received on or before this date.

ADDRESSES: Submit comments to: Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Service Branch. For a copy of the petition, write: Rules Review Section, Regulatory Publications Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

FOR FURTHER INFORMATION CONTACT: Michael T. Lesar, Chief, Rules Review Section, Regulatory Publications Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Telephone: 301 492-7758 or Toll Free: 800-368-5642.

SUPPLEMENTARY INFORMATION:

Petitioners' Request

The petitioners request that the Commission amend 10 CFR 60.2 to clarify the definition of "high-level radioactive waste" (HLW) and the definition of "HLW facility." The petitioners request that the Commission—

1. Establish a process to evaluate the treatment of defense reprocessing wastes in tanks so that such wastes will not be considered HLW if, prior to disposal, each tank is treated to remove the largest technically achievable amount of radioactivity; and

2. Require that the heat produced by residual radionuclides, together with the heat of reaction during grout processing (if employed as a treatment technology), will be within limits established to ensure that grout meets temperature requirements for long-term stability for low-level waste forms.[1]

The petitioners seek clarification that the disposal of wastes treated to this standard is not disposal in a "HLW facility" as presently defined in 10 CFR 60.2. The petitioners state that should the Commission regard 10 CFR Part 50, Appendix F as the controlling regulation to determine whether a waste is HLW, that the Commission also modify that definition as proposed in the petition.

Basis for the Petition

The petitioners state that this rulemaking is based, in part, on section 202 of the 1974 Energy Reorganization Act, which defines Commission authority over retrievable surface storage facilities and other facilities authorized for the express purpose of subsequent long-term storage of high-level radioactive waste generated by DOE which are not used for, or are part of, research and development activities. The petitioners further state that the Congressional definition of the term "high-level radioactive waste" in the Nuclear Waste Policy Act (NWPA) 42 U.S.C. 10101 (12) gives the Commission the authority to define whether wastes are "highly radioactive material" or "solids derived from [liquid reprocessing wastes] that contain fission products in sufficient concentrations."[2]

According to the petitioners, legislative history reveals that Congress intended the Commission to license defense reprocessing tank wastes at the point of long-term storage or disposal. The petitioners note that low fraction wastes resulting from pretreatment of tank wastes are scheduled to be grouted and disposed of in land-based grout vaults on the Hanford site in accordance with regulations developed under the Resource Conservation and Recovery Act (RCRA). The petitioners believe that if such wastes are HLW, they clearly fall under the Commission's licensing jurisdiction under section 202 (4) of the Energy Reorganization Act of 1974.[3]

Reasons for Petition

The petitioners point out that the present definition of HLW in the Commission's regulations is based upon the source of the waste. According to petitioners, while HLW may be differentiated from "incidental waste," the legal basis for doing so must derive from NWPA, specifically 42 U.S.C. 10101 (12) (A), which refers to a "sufficient concentrations" criterion for classification.[4] The petitioners claim that incidental waste source is impossible to ascertain due to mixing in defense tanks and the unavailability of accurate records. They point out, in particular, that over the last 45 years, mixing of wastes from different sources has complicated the classification of Hanford tank wastes, including double-shell tank wastes. Moreover, the petitioners state that radionuclide inventories are estimates and subject to substantial uncertainty. Variables contributing to the uncertainty include incomplete and inaccurate records, the lack of actual fuel and/or waste analyses, and an incomplete understanding of the chemistry and

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[1] Grout is a fluid mixture of cementitious materials and liquid waste that sets up as a solid mass and is used for waste fixation and immobilization.

[2] For an analysis of this provision, see "Definition of 'High-Level Radioactive Waste'" (advance notice of proposed rulemaking, 52 FR 5992, February 27, 1987) and subsequent rulemaking documents (proposed amendments to 10 CFR part 61, 53 FR 17709, May 18, 1988; final amendments to 10 CFR part 61, 54 FR 22578, May 25, 1989).

[3] It should be noted, however, that the Commission has jurisdiction only if the facilities are of the types described in section 202(4).

[4] Note, however, the Commission's statement, at 52 FR 5993, February 27, 1987, that classification under the cited provision "would be irrelevant in determining whether such wastes must be disposed of in licensed disposal facilities."

pathways in reprocessing and waste treatment processes. The petitioners assert that neither DOE, the Commission, nor the petitioners have adequate information regarding the radioactive portion of the double-shell tank waste. The petitioners believe that "the Commission needs to establish both a procedure and a standard for making an evaluation as to whether waste are HLW on a tank-by-tank basis.

The petitioners assert that the proposed amendment is essential to provide protection of the future health and safety of the citizens of the Pacific Northwest.

Petitioners' Proposal

The petitioners suggest that the definitions of "High-Level Radioactive Waste" and "HLW Facility" in 10 CFR 60.2 be revised and a new appendix A be added to 10 CFR part 60. The specific language suggested by the petitioners reads as follows:

1. In § 60.2, the definitions of "High-Level Radioactive Waste" and "HLW Facility" are revised to read as follows:

§ 60.2 Definitions.

* * * * *

High-level radioactive waste or HLW means: (1) Irradiated reactor fuel, (2) Liquid wastes resulting from the operation of the first cycle solvent extraction system, or equivalent, and the concentrated wastes from subsequent extraction cycles, or equivalent, in a facility for reprocessing irradiated reactor fuel, and (3) Solids into which such liquid wastes have been converted; provided that if, prior to disposal, defense reprocessing tank wastes are treated to remove the largest technically achievable amount of radioactivity on a tank-by-tank basis (as provided in appendix A), the treated residual fraction shall be considered an incidental waste and therefore not HLW.

HLW facility means a facility subject to the licensing and related regulatory authority of the Commission pursuant to sections 202(3) and 202(4) of the Energy Reorganization Act of 1974 (88 Stat 1244).[8]

* * * * *

[8] These are DOE "facilities used primarily for the receipt and storage of high-level radioactive wastes resulting from activities licensed under such Act (the Atomic Energy Act)" and "Retrievable Surface Storage Facilities and other facilities authorized for the express purpose of subsequent long-term storage of high-level radioactive waste generated by [DOE], which are not used for, or are part of, research and development activities". Facilities for the long-term storage or disposal of incidental wastes resulting from treatment of defense reprocessing wastes are not HLW facilities.

2. A new Appendix—A is added to part 60 to read as follows:

Appendix A—Procedures For Determining Largest Technically Achievable Treatment

At least one year before a tank of defense reprocessing wastes containing high-level waste components is treated, pretreated or blended prior to permanent disposal, DOE shall submit the following to the Commission and the affected state and publish in the Federal Register:

1. Data on physical characteristics of the waste, including density and percent solids, inorganic and organic constituents, and radiochemistry (e.g., gamma energy analysis, total alpha, total beta);

2. Volumetric data on untreated waste, on volume changes expected as a result of treatment, pretreatment or blending activities and the expected volume of the final waste form (grout, saltcrete or vitrified waste);

3. A description of the treatment processes, including an estimated mass balance for each process, and estimated percent recovery for each separation, and concentrations of major waste components before and after treatment;

4. The proposed grout or saltcrete formulation, together with heat transfer calculations for the waste form; and

5. To the degree possible, treatment system models similar to the attached grout system model should be used to present data and describe processes.

At least six months before a tank of defense reprocessing tank wastes containing high-level waste components is pretreated, treated or blended prior to permanent disposal in near-surface or deep geologic facilities, the Commission shall require a license under section 202(4) of the Energy Reorganization Act, 42 U.S.C. 5842 (4) unless the Commission, on a tank-by-tank basis determines the following:

1. The DOE has demonstrated that the largest technically achievable amount of activity from the tank will be isolated for vitrification prior to permanent disposal; and

2. That use of permanent shallow land disposal for the tank waste will be limited to the incidental waste portion, which is the activity remaining after the largest technically achievable amount of activity has been removed; and

3. That the treatment, pretreatment and blending processes described in the DOE submittal will achieve the stated separation and/or recovery efficiencies; and

4. That the treatment, pretreatment and blending processes described in the DOE submittal are proven, cost effective, state-of-the art processes, which are capable of removing the largest technically achievable amount of activity.

Petitioners' Conclusions

The petitioners state that rulemaking procedures are necessary to determine the nature of the incidental, lesser radioactive fraction of wastes and that rulemaking is appropriate to establish a procedural framework and substantive standards by which particular wastes will be assessed. The petitioners contemplate that particular determinations of how specific waste will be characterized under these general standards can be left to individual adjudicative proceedings.

The petitioners believe that the amendments suggested by their petition would protect human health and the environment, would facilitate meaningful Commission involvement in the ultimate disposal and/or long-term storage of Hanford double-shell tank waste, and would support implementation of the Hanford Federal Facility Agreement and Consent Order.

Request for Comments

Commenters are invited to address, among other things, the desirability and appropriateness of (1) The proposed substantive standard ("remove the largest technically achievable amount of radioactivity on a tank-by-tank basis"), (2) the proposed procedure for applying that standard, and (3) an amendment to 10 CFR part 60 (in view of the scope defined in 10 CFR 60.1) vis-a-vis the adoption of a new Part or amendment to some other existing Part of NRC regulations.

Dated at Rockville, Maryland, this 11th day of December 1990.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,

Secretary of the Commission.

[FR Doc. 90-29430 Filed 12-14-90; 8:45 am]

BILLING CODE 7590-01-M

DEPARTMENT OF COMMERCE

Foreign-Trade Zones Board

15 CFR Part 400

[Docket No. 21222-0262]

RIN 0625-AA84

Foreign-Trade Zones in the United States

AGENCY: Foreign-Trade Zones Board, International Trade Administration, Commerce.

ACTION: Notice of proposed rulemaking; extension of comment period.

SUMMARY: In response to requests from interested parties, the period for public comment on the further amendments to the proposed revisions to the regulations of the Foreign-Trade Zones Board regarding foreign-trade zones in the United States published in the Federal Register on November 20, 1990 (55 FR 48446), is extended to February 1, 1991.

DATES: Comments must be received on or before February 1, 1991.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

MAR 0 1 1993

Ms. Jill Lytle
Deputy Assistant Secretary for Waste Operations
Office of Waste Management
Environmental Restoration
and Waste Management
U.S. Department of Energy
Washington, D.C. 20585

Dear Ms. Lytle:

Members of the Nuclear Regulatory Commission staff appreciated the opportunity to meet with the Department of Energy (DOE) staff, DOE contractors, and other parties on July 16, 1992, to review new waste characterization data and current DOE plans for management of radioactive tank waste at Hanford. The purpose of this letter is to provide DOE with the staff's assessment of that information as it relates to DOE's program to classify, process, and dispose of Hanford tank wastes. We are also taking this opportunity to respond to the related November 4, 1992, letter from Leo P. Duffy to Chairman Ivan Selin.

During the meeting, DOE presented revised tank waste inventory estimates, based on current characterization data. The information indicated that the double-shell tank activity that would be grouted in near-surface vaults is within earlier range estimates. The NRC staff is concerned, however, that Cs-137 quantities are now near the upper end of the range, rather than at the lower end, as previously believed, especially given that DOE indicated that uncertainties associated with the activity estimates remain because of the limited sampling and analysis that has been conducted to date. Consequently, we encourage DOE to examine available mechanisms for achieving greater radionuclide separation.

In presenting its current plans for waste management, DOE outlined its intention to complete, by March 1993, a broad reevaluation of various treatment options for both single and double-shell tanks. These options include a new facility to be used to separate radionuclides for repository disposal of high-level radioactive waste (HLW).

As you recall, NRC indicated to DOE, in 1989, its agreement that the criteria DOE used for classification of grout feed as low-level waste were appropriate, and, consequently, that the grout facility for disposal of double-shell tank

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waste would not be subject to our licensing authority (R. Bernero, NRC letter to A. Rizzo, DOE, September 25, 1989). This agreement was predicated on our understanding that DOE would segregate the largest practical amount of the total site activity attributable to "first-cycle solvent extraction, or equivalent" for disposal as HLW, leaving behind only a small fraction of moderately radioactive material.

The Commission has recently completed its review of a rulemaking petition from the States of Washington and Oregon on the subject of the double-shell tank wastes and has indicated, in the enclosed petition denial, that it would regard the residual fraction as "incidental" waste, based on the Commission's understanding that DOE will assure that the waste: (1) has been processed (or will be further processed) to remove key radionuclides to the maximum extent that is technically and economically practical; (2) will be incorporated in a solid physical form at a concentration that does not exceed the applicable concentration limits for Class C low-level waste as set out in 10 CFR Part 61; and (3) will be managed, pursuant to the Atomic Energy Act, so that safety requirements comparable to the performance objectives set out in 10 CFR Part 61 are satisfied.

It is therefore essential, in the light of this position, that DOE's present reevaluation of tank waste remediation options, and subsequent periodic evaluations as may be conducted, include the application of these principles. We recognize that there may be significant economic, programmatic, and safety factors affecting the remediation program, but the consideration of such factors as they may relate to the possible jurisdiction of NRC should be made clear.

If, during your periodic evaluations, it becomes apparent to you that any wastes may be subject to NRC licensing, it will be necessary for you to communicate that concern to NRC. It will then be necessary to determine what form of pre-licensing interactions, analogous to repository site characterization, would be needed to define the appropriate disposition of these wastes. We expect that DOE will document the results of the analyses supporting its conclusions and that this documentation will be adequate for an NRC review, should that be appropriate. We believe it would be prudent for any such documentation to be developed with good record-keeping and under an adequate quality assurance process.

I trust that this letter and the enclosed petition denial provide the information requested in Leo P. Duffy's November 4, 1992, letter to Chairman Ivan Selin, regarding NRC's intended response to the rulemaking petition by

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the States of Washington and Oregon. If you have any further questions, please feel free to contact me, at 301-504-3352, or B.J. Youngblood, Director of the Division of High-Level Waste Management, at 301-504-3404.

Sincerely,

Robert M. Bernero, Director
Office of Nuclear Material Safety
and Safeguards

Enclosure
Petition Denial

cc. J. Tseng, DOE-EM-36
J. Anttonen, DOE
L. Barrett, DOE-RW-1
P. Grimm, DOE-EM-1
D. Duncan, EPA
R. Stanley, Washington State
J. Franco, Oregon State
R. Jim, YIN

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Dated at Rockville, Maryland, this 26th day of February 1993.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,

Secretary of the Commission.

[FR Doc. 93-4543 Filed 3-3-93; 8:45 am]

BILLING CODE 7590-01-P

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10 CFR Part 60

[Docket No. PRM-60-1]

States of Washington and Oregon; Denial of Petition for Rulemaking

AGENCY: Nuclear Regulatory Commission.

ACTION: Denial of petition for rulemaking.

SUMMARY: The Nuclear Regulatory Commission (NRC) is denying a petition for rulemaking (PRM-60-1), submitted by the States of Washington and Oregon, which deals with the process and criteria for classifying radioactive waste materials at defense facilities as high-level radioactive waste (HLW) or as non-HLW. (As noted in the petition, certain facilities for the storage of HLW are subject to NRC licensing authority.) The petition is being denied because the NRC concludes that the principles for waste classification are well established and can be applied on a case-by-case basis without revision to the regulations.

ADDRESSES: Copies of the petition for rulemaking, the public comments received, and the NRC's letter to the petitioner are available for public inspection or copying in the NRC Public Document Room, 2120 L Street, NW. (Lower Level), Washington, DC.

FOR FURTHER INFORMATION CONTACT: Naiem S. Tanious, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone (301) 492-3878.

SUPPLEMENTARY INFORMATION:

I. The Petition

The States of Washington and Oregon, and the Yakima Indian Nation, initially submitted a petition for rulemaking on this subject on January 2, 1990. On February 7, 1990, the NRC staff conferred with the petitioners as contemplated by paragraph (b) of 10 CFR 2.802. In response to suggestions by the NRC staff, the petition was clarified and resubmitted (by the States of Washington and Oregon) on July 27, 1990.

On December 17, 1990, the Nuclear Regulatory Commission published a notice of receipt of the petition for rulemaking (55 FR 51732). The petition requested that the Commission revise the definition of "high-level radioactive waste" (HLW) so as to establish a procedural framework and substantive standards by which the Commission will determine whether reprocessing waste, including in particular certain waste stored at the U.S. Department of Energy's (DOE) site at Hanford, Washington, is HLW and, therefore, subject to the Commission's licensing authority.

The petitioners request that the Commission amend 10 CFR 60.2 to clarify the definition of HLW and the definition of "HLW facility." The petitioners specifically request that the Commission:

1. Establish a process to evaluate the treatment of defense reprocessing wastes in tanks so that such wastes will not be considered HLW if, prior to disposal, each tank is treated to remove the largest technically achievable amount of radioactivity; and

2. Require that the heat produced by residual radionuclides, together with the heat of reaction during grout processing (if employed as a treatment technology), will be within limits established to ensure that grout meets temperature requirements for long-term stability for low-level waste forms.

The petitioners state that the petition for rulemaking is based, in part, on Section 202 of the Energy Reorganization Act of 1974 (ERA), which provides for the Commission to exercise licensing and related regulatory authority over "facilities authorized for the express purpose of subsequent long-term storage of high-level radioactive wastes generated by [DOE] which are not used for, or are part of, research and development activities."

According to the petitioners, the legislative history of the ERA reveals that Congress intended the Commission to license defense reprocessing tank wastes at the point of long-term storage or disposal. The petitioners note that "low-fraction wastes" resulting from pretreatment of tank wastes are scheduled to be grouted and disposed of in land-based grout vaults on the Hanford site in accordance with regulations developed under the Resource Conservation and Recovery Act (RCRA). The petitioners believe that if these wastes are HLW, they clearly fall under the Commission's licensing jurisdiction under Section 202(4) of the Energy Reorganization Act of 1974 (42 U.S.C. 5842(4)).

The petitioners acknowledge that the present definition of HLW in the Commission's regulations is based upon the source of the waste, and that "incidental waste" generated in the course of reprocessing is not HLW. (The latter point is evident from the proposal to amend 10 CFR 60.2 to provide that a residual fraction would be "considered an incidental waste and, therefore, not HLW.") The petitioners claim, however, that wastes stored in tanks at Hanford cannot precisely be classified as incidental waste (as opposed to HLW) because the tanks contain a mixture of wastes from a number of sources, including reprocessing of reactor fuel. Moreover, the petitioners note that radionuclide inventories are estimates subject to substantial uncertainty, owing to lack of accurate records. Further, the petitioners assert that neither DOE, the Commission, nor the petitioners have adequate information regarding the source and composition of the tank waste. Hence, the petitioners believe that the Commission needs to establish both a procedure and a standard for making an evaluation as to whether wastes are HLW on a tank-by-tank basis.

The petitioners assert that the proposed amendment is essential to provide protection of the future health and safety of the citizens of the Pacific Northwest.

II. Classification of DOE Reprocessing Wastes

At Hanford and other sites, questions have arisen regarding the classification of reprocessing wastes for which DOE must provide disposal. In the long-standing view of the Commission, these questions must be resolved by examining the source of the wastes in question. The reason for this is that when Congress assigned to NRC the licensing authority over certain DOE facilities for "high-level radioactive wastes," the Congress was referring to those materials encompassed within the meaning of the term "high-level radioactive waste" in Appendix F of 10 CFR Part 50. (For a full statement of this position, see the discussion presented in the Commission's advance notice of proposed rulemaking, "Definition of High-Level Radioactive Waste" (52 FR 5993, February 27, 1987).) Accordingly, any facility to be used for the disposal of "those aqueous wastes resulting from the operation of the first cycle solvent extraction system, or equivalent . . ." as HLW is defined in Appendix F to Part 50, must be licensed by the NRC. Most of the waste storage tanks at Savannah River (South Carolina), West Valley (New York), and Hanford contain wastes that meet this definition, and the facilities to be used for disposal of these wastes are, therefore, potentially subject to NRC licensing jurisdiction.

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However, when the Appendix F definition was promulgated, the Atomic Energy Commission specifically noted that the term HLW did not include "incidental" waste resulting from reprocessing plant operations, such as ion exchange beds, sludges, and contaminated laboratory items, such as clothing, tools, and equipment. Neither were radioactive hulls and other irradiated and contaminated fuel structural hardware encompassed by the Appendix F definition. Under the same reasoning, as the Commission has previously indicated, incidental wastes generated in further treatment of HLW (e.g., salt residues or miscellaneous trash from waste glass processing) would be outside the Appendix F definition.

In the cases of Savannah River and West Valley wastes, DOE plans to retrieve the wastes from their storage tanks and to separate essentially all of the radioactive materials for eventual disposal in a deep-geologic HLW repository. Accordingly, the projected recovery of HLW from the wastes in tank storage at those sites will be sufficiently complete that the decontaminated salts and other residual wastes are classified as "incidental" (i.e., non-HLW). The NRC will have no regulatory authority, under Section 202 of the Energy Reorganization Act, over DOE's facilities to be used for processing and disposal of the incidental waste.

At Hanford, DOE plans to process the wastes presently stored in double-shell tanks in a manner similar to that planned for the wastes at Savannah River and West Valley. Such processing would separate most of the radioactive constituents of the wastes for eventual deep-geologic repository disposal and, the residual salts would be disposed of onsite in a shallow, near-surface concrete-like grout facility. (Plans for processing of single-shell tank wastes have been deferred.) However, classification of the Hanford double-shell tank wastes has proven more difficult than classification of Savannah River and West Valley wastes. At Hanford, many of the primary reprocessing wastes were generated using older separation technologies, which resulted in substantial dilution of those wastes with nonradioactive materials. In addition, many of the tanks at Hanford contain mixtures of wastes from both reprocessing sources and other sources. Finally, recordkeeping at Hanford was not always thorough enough to allow precise determinations of the origins of the wastes now present in specific tanks at Hanford. For these reasons, some of the Hanford tank wastes cannot be readily classified as either HLW or incidental wastes using only the definitions and concepts discussed above.

Taking into account these uncertainties and their implications with respect to NRC jurisdiction, the NRC and DOE staff held several meetings to explore the situation in detail. A principal objective of these meetings was to ascertain, to the extent practicable, whether some or all of the wastes should be regarded as HLW and whether, on the other hand, some or all of the wastes should be classified as non-HLW. Several things became clear as a result of these meetings.

First, management records were adequate for DOE to determine that two double-shell waste tanks do not contain wastes from reprocessing of reactor fuels. Therefore, these wastes clearly do not contain HLW within the Appendix F definition. The NRC agreed with DOE that any disposal facility intended exclusively for these wastes would not be subject to NRC licensing authority.

Second, DOE has carried out a "material balance" analysis of waste management activities at Hanford. This analysis estimated the total amount of "first cycle reprocessing wastes" generated at Hanford and, to the extent practical, the current location of those wastes. The DOE proposed onsite grout disposal of the residual waste from the double-shell tank waste processing would be only a small fraction of the reprocessing wastes originally generated at the site.

Finally, DOE studied possible technologies for additional waste processing, and agreed to remove the largest practical amount of radioactive material from double-shell tank wastes prior to disposal in onsite grout facilities. This commitment by DOE, coupled with the material-balance study indicating that most of the originally-generated radioactive material would be recovered, led the NRC staff to conclude that the residual waste material should be classified as incidental waste, since they are wastes incidental to the process of recovering HLW. With this classification, DOE could proceed with onsite disposal of such incidental wastes in a grout facility without licensing by the NRC. It should be noted that if the DOE processing operations go as planned, the residual activity of these incidental wastes would be below the concentration limits for Class C wastes under the waste classification criteria of 10 CFR part 61.

Following its review, the NRC staff, by letter dated September 25, 1989, from R.M. Bernero, Director, Office of Nuclear Material Safety and Safeguards, NRC, to A.J. Rizzo, Assistant Manager for Operations, Richland Operations Office, DOE, endorsed DOE's plans to sample and analyze the grout feeds before disposal in an effort to control the final composition of the grout feed. However, the staff indicated that if DOE were to find, in the course of conducting the sampling program, that the inventories of key radionuclides entering the grout facility are significantly higher than previously estimated, DOE should notify the NRC and other affected parties in a timely manner.

It should be noted that the appropriate classification of some Hanford wastes remains to be determined—specifically, any single-shell tank wastes, and any empty but still contaminated waste tanks DOE might dispose of in-place. For both types of wastes, a case-by-case determination of the appropriate waste classification might be necessary.

III. Discussion

The petition for rulemaking presents two basic issues. The question is not whether "high-level waste" should be interpreted by reference to the source-based concepts derived from appendix F to 10 CFR part 50. The petitioners agree that this is proper. Nor is there any fundamental challenge to the concept that "incidental wastes" are excluded from the definition of "high-level waste." The issues are much narrower ones. The first issue is a substantive one—the criteria to be applied in differentiating incidental waste from high-level waste. The second issue is a procedural one—the process that should be employed by the Commission in arriving at a judgment whether or not it has jurisdiction over particular facilities. These will be addressed in turn.

A. The Standard for Classification

We first address the standard that should be employed in distinguishing high-level waste from incidental waste. In doing so, we strive to apply the policies that underlie the adoption of appendix F to 10 CFR part 50 (and, hence, section 202 of the Energy Reorganization Act)

[1] See 52 FR 5993, February 27, 1987 (definition of "high-level waste"), n. 1, where the Commission characterizes as "incidental waste," the decontaminated salt with residual activities on the order of 1,300 nCi/g Cs-137, 30 nCi/g Sr-90, 3 nCi/g Pu, as described in the Department of Energy's FEIS on long-term management of defense HLW at the Savannah River Plant, DOE/EIS-0023, 1979. Although an EIS has not yet been published for the West Valley Demonstration Project, preliminary estimates indicate the likelihood of an equivalent degree of separation.

The petitioners suggest that the proper standard, to be applied on a tank-by-tank basis, is to consider all processing streams to be high-level waste unless they have been treated, prior to disposal, "to remove the largest technically achievable amount of radioactivity." Adoption of such a criterion would certainly serve the goal, which had been contemplated by the Commission, of removing the hazardous process streams to a geologic repository for permanent storage. It is not the only standard, however, that would suffice for this purpose, particularly when it is viewed in a broader regulatory context.

The clearest expression of the overall regulatory objectives is the Atomic Energy Commission's (AEC's) explanatory statement when it promulgated appendix F—namely, "that the public interest requires that a high degree of decontamination capability be included in such facilities and that any residual radioactive contamination after decommissioning be sufficiently low as not to represent a hazard to the public health and safety." 35 FR 17530, November 14, 1970. As we read the AEC's intent, the reference to "a high degree of decontamination capability" leaves a substantial degree of discretion. It certainly does not rule out consideration of economic factors as well as technical ones. It was the AEC's contemporaneous practice to consider financial impacts as, for example, in controlling releases of radioactive materials from licensed facilities to the lowest levels "technically and economically practical." AEC Manual Chapter 0511. When the AEC spoke of a "high degree" of decontamination capability, we believe that it was guided by similar considerations. Moreover, from a policy standpoint, this makes good sense, for so long as there is adequate protection of public health and safety, it would not be prudent to expend potentially vast sums without a commensurate expectation of benefit to health and the environment.

Achieving a "high degree of decontamination capability" implies, then, that the facility should separate for disposal as much of the radioactivity as possible, using processes that are technically and economically practical. In addition, however, as the AEC's statement indicates, the residual radioactive contamination should be sufficiently low as not to endanger public health and safety.

These principles—high decontamination capability and protection of health and safety—are the essential benchmarks that have influenced the development of NRC's position vis-a-vis DOE on the question of the proper classification of the tank wastes and grout at Hanford.

When the question regarding classification of wastes was first raised, the NRC staff identified to DOE some approaches that might be used in distinguishing HLW from incidental waste. One approach was expressed as follows: [2]

As an alternative approach, we suggest that DOE attempt an overall material balance for HLW at the Hanford site, using the source-based meaning of HLW. It is hoped that this approach might provide a more efficient means of identifying those wastes subject to licensing by NRC under terms of the 1974 Energy Reorganization Act. Under this approach, if DOE could demonstrate that the largest practical amount of the total site activity attributable to "first-cycle solvent extraction" wastes has been segregated for disposal as HLW, then NRC would view the residual as a non-HLW. We would anticipate that at least 90 percent of the activity would have been separated in this way. Thus, if it can be shown that DOE has processed the waste with the intent to dispose of the HLW in a repository or other appropriate licensed facility, leaving behind only a small fraction of only moderately radioactive material, then the goals stated in 10 CFR part 50 appendix F and incorporated in the Energy Reorganization Act would have been satisfied; and the disposal of the residual would accordingly not be subject to NRC licensing.

In response, DOE considered the practicality of various waste processing alternatives and presented the results of its study by letter dated March 6, 1989. [3] The results were also presented at a meeting among interested parties, including the petitioners, held on August 4, 1989. (Minutes of the meeting are available for public inspection in the NRC Public Document Room) DOE's "baseline" disposal plans would have recovered all but about 12–13 million curies of cesium-137, together with lesser activities of strontium-90, transuranics, and other radionuclides. [4] DOE's study indicated the practicality of removing an additional 6 million curies of cesium-137 for repository disposal. DOE proposed to remove this additional 6 million curies of cesium-137. DOE also identified additional treatment alternatives, with their associated costs which it viewed as not being economically practical. DOE's material balance showed that, after the residue from the double-shell tank wastes is grouted, 2 to 3 percent of the key radionuclides which originally entered all Hanford tanks would be disposed of as LLW in near-surface vaults. The concentrations of radionuclides in the grout would be comparable to Class C for cesium and transuranic wastes, and to Class A or B for the remainder. [5] DOE also noted certain engineering and institutional factors that might compensate, especially as to potential intrusion hazards, for the possibility that the total amount of waste that would be grouted would be greater than the amount of Class C waste that might be contained in a typical commercial burial ground.

Based on its review of DOE's March 6, 1989 submission, the NRC staff concluded that DOE's proposed processing would remove the largest practical amount of total site activity, attributable to HLW, for disposal in a deep geologic repository. This finding was based on: (1) Past and planned treatment of the tank wastes; (2) radionuclide concentration and material balance; and (3) cost-effectiveness of additional radionuclide removal. These conclusions reflected DOE's undertakings both to achieve a high degree of separation and to provide protection of public health and safety. As a result, the staff concluded that the expected residual waste would not be high-level waste and would thus not be subject to NRC licensing authority. The staff thereupon advised DOE that NRC agreed that the criteria used by DOE for classification of the grout feed are appropriate and that the grout facility for the disposal of the double-shell tank waste would not be subject to NRC licensing authority. [6]

[2] Letter from Michael J. Bell, Chief, Regulatory Branch, Division of Low-Level Waste Management and Decommissioning, Office of Nuclear Material Safety and Safeguards, NRC, to Ronald E. Gerton, Director, Waste Management Division, Richland Operations Office, DOE, November 29, 1988. The letter included some "suggested criteria" involving a "good faith" effort to achieve isolation of HLW from nonradioactive salts, such an effort to be judged, on a practical manner, by considering (among other things) alternative separation processes.

[3] Letter from A. J. Rizzo, Assistant Manager for Operations, Richland Operations Office, DOE, to Robert M. Bernero, Director, Office of Nuclear Materials Safety and Safeguards, NRC, March 6, 1989.

[4] DOE noted in the March 6, 1989 letter from Rizzo to Bernero that, based on limited available analytical data, the total cesium-137 could be as much as 30 million curies versus the 12–13 million estimate.

[5] NRC understood this statement to connote that cesium-137 and transuranic radionuclides in the residual waste would be less than the concentration limits for Class C low-level waste, as defined in NRC's requirements in 10 CFR Part 61, and that the concentrations of other radionuclides would be less than the concentration limits for Class A or B low-level waste.

[6] Letter from Robert M. Bernero, Director, Office of Nuclear Material Safety and Safeguards, NRC, to A. J. Rizzo, Assistant Manager for Operations, Richland Operations Office, DOE, September 23, 1989. The letter also called upon DOE to advise NRC periodically of the analytical results of samples of key radionuclides entering the grout facility, so that the classification of the waste might be reconsidered if the inventories were significantly higher than DOE had estimated.

At a meeting in Richland, Washington on July 16, 1992, DOE staff presented more detailed double-shell tank waste processing options and, based on recent analyses, summarized available information on the characteristics of waste within the tanks. DOE's current estimate of the total amount of radioactivity proposed for disposal in grout in near-surface vaults is within earlier range estimates but is now believed to be nearer the upper end of the range. DOE also clarified its intention to apply criteria comparable to the Performance Objectives set out in 10 CFR part 61. Among other things, these performance objectives include numerical radiation exposure limits for protection of the general population from releases of radioactivity and requires a design to achieve long-term stability of the disposal site.

DOE intends to complete a reassessment of the tank waste processing options by March 1993. This reassessment, the NRC staff understands, will include a reexamination of the practicality of achieving higher degrees of separation, particularly with respect to those tanks that contain substantial quantities of key radionuclides.

Assuming implementation of DOE's plans as described above, the Commission concludes that any radioactive material from the double shell tanks that is deposited in the grout facility would not be high-level radioactive waste subject to NRC's licensing jurisdiction. The responsibility for safely managing those wastes rests with the Department of Energy. The basis for the Commission's conclusion is that the reprocessing wastes disposed of in the grout facility would be "incidental" wastes because of DOE's assurances that they: (1) Have been processed (or will be further processed) to remove key radionuclides to the maximum extent that is technically and economically practical; (2) will be incorporated in a solid physical form at a concentration that does not exceed the applicable concentration limits for Class C low-level waste as set out in 10 CFR part 61; and (3) are to be managed, pursuant to the Atomic Energy Act, so that safety requirements comparable to the performance objectives set out in 10 CFR part 61 are satisfied.

The petitioners also requested that the Commission exercise oversight to assure that the grout meets temperature requirements for low-level waste forms. They acknowledge that DOE's vault design is protective of human health and the environment if heat produced by residual radioactivity, together with heat generated from reactions during the grout process, is kept within defined limits. They present no technical data to suggest that achievement of these temperature controls presents any unusual engineering challenge. In any event, inasmuch as the Commission does not consider the grout producing in accordance with DOE's plans to be high-level waste, it does not have the authority to carry out this oversight function.

B. Procedural Issues

1. Whether Rulemaking Is Necessary and Desirable

The petitioners urge that the Commission initiate rulemaking procedures that would result in the establishment of substantive criteria for determining whether particular radioactive wastes either are or are not high-level waste. Generally, a decision whether to proceed by rulemaking (as requested) or to make determinations in individual, ad hoc litigation lies within the informed discretion of the cognizant administrative agency. Rulemaking is most appropriate where an agency seeks to establish a general principle, having prospective effect, to be applied in a wide variety of factual contexts. Where the issue before an agency involves the application of law to a very specific existing fact situation, especially where that situation is not representative of other matters that may need to be decided by the agency, then it is clearly more efficient and more to the point to decide by a process of adjudication (i.e., on a case-by-case basis).

Applying these principles to the petition at hand, the Commission has little difficulty in concluding that rulemaking is neither necessary nor desirable. Reprocessing wastes are located at only four principal locations in the United States. The Commission has previously determined that the residual contamination anticipated from proposed operations at Savannah River should be characterized as incidental waste and not high-level waste (see 52 FR 5993, Feb. 27, 1987, cited above, at footnote 1.) Wastes generated at the Idaho Chemical Processing Plant are markedly different from those at Hanford and Savannah. Therefore, if questions about classification of the Idaho wastes should arise, precedents established at Savannah River and Hanford might be difficult to apply. Any wastes at the Western New York Nuclear Service Center will require treatment in accordance with the applicable provisions of the West Valley Demonstration Project Act.

The limited practical effect of the decision—i.e., restricted to the Hanford tanks—is reason enough to proceed by way of adjudication instead of rulemaking. The Commission is persuaded further by the need to avoid making premature decisions with respect to the wastes stored at Hanford in single-shell tanks that are not the subject of pending treatment plans. If the Commission were to establish rules to apply to the wastes remaining in those tanks, our inquiry would have to be greatly broadened; and it might become necessary to consider a wide range of situations that might or might not ever come to pass in the future.

2. Whether the Commission Is Adequately Informed

Petitioners suggest that their proposed procedures, which include detailed tank-by-tank assessment, are necessary to ensure confidence in the treatment process employed by DOE and to build confidence that the treatment standard is being met.

The issue to be decided by the Commission is a much narrower one: It is merely to determine whether the activities being undertaken by the Department of Energy fall within the NRC's statutory jurisdiction. As in the case of other persons whose activities may fall within our regulatory sphere, the Commission may from time to time demand information so as to be able to determine whether or not to initiate an enforcement action. The NRC staff has acted in this manner in its inquiries to DOE. It has obtained and evaluated information that is relevant and material to a determination whether or not the proposed activities of the DOE are subject to NRC licensing jurisdiction. All the information obtained and evaluated has been made available contemporaneously to the public.

Moreover, as a practical matter, NRC recognized the uncertainties associated with the projected radionuclide inventories in the tank wastes and endorsed DOE plans for sampling and analyzing the grout feeds before disposal. The objective of these efforts is to control the final composition of the grout wastes. If DOE finds that it can no longer assure that these wastes will be managed in accordance with the criteria previously discussed, DOE should notify NRC.

If a standard of "largest technically achievable amount * * * will be isolated" were to be applied, then the facts submitted by DOE might not be sufficient to conclude that NRC lacked jurisdiction. However, the proper standard includes considerations of economical practicality as well. As

indicated in an earlier part of this decision, the Commission has obtained information that is sufficient for this purpose.

3. Future Adjudications

The petitioners contemplate that if a rule were to be adopted in accordance with their proposal, particular determinations of how specific wastes would be characterized would be "left to individual adjudicative proceedings." The NRC infers that the "proceedings" contemplated by petitioners are licensing activities of the kinds specified in Section 189 of the Atomic Energy Act, as amended, 42 U.S.C. 2239. Adjudications in this type of proceeding are in some cases to be conducted in accordance with the hearing provisions of subpart L of 10 CFR part 2.

These procedures are often appropriate with respect to activities that are subject to NRC regulatory and licensing authority. However, the NRC is reluctant to employ them in the context that is proposed—to determine whether NRC has jurisdiction in the first place. To do so would entail the conduct of an adjudicatory proceeding in order to see whether another adjudicatory licensing proceeding must be held. More importantly, the Commission considers that the existing record contains all the factual information needed for a decision and that no unresolved material factual issues remain that would require further proceedings.

4. Other Considerations

While both NRC and DOE have focused their attention upon the meaning of the statutory term "high-level waste" and its application to the materials in storage at Hanford, other considerations might come into play in determining whether or not DOE activities are subject to licensing. In particular, it should be recalled that NRC exercises licensing authority under section 202(4) only as to "facilities authorized for the express purpose of subsequent long-term storage of [DOE-generated] high-level waste." The content of individual waste tanks is by no means dispositive of the question whether the facilities for storage of the treated waste are subject to licensing. A number of other factors may be relevant and material as well: (1) What are the limits, geographically and functionally, of "facilities"; (2) have these facilities been "authorized" (and by whom is such authorization required); and (3) have these facilities been authorized "for the express purpose of subsequent long-term storage of high-level waste" where those who may authorize the facility make no express mention of high-level waste? It is not necessary for the Commission to address these questions at length in order to dispose of the pending petition.

IV. Public Comments on the Petition

The NRC received letters from 12 commenters. Two letters were from other Federal agencies, two were from public interest groups, one was from a nuclear industry corporation, and seven were from private individuals. Most comments were opposed to the petition.

A. Process and Standards Proposed in Petition

Several comments expressed concern that granting the petition would have an adverse effect on the timely disposal of radioactive waste at Hanford. This was a concern because many of the Hanford waste tanks were seen as nearing or exceeding their design life. The provisions of the rulemaking proposed in the petition were viewed as limiting DOE's flexibility in selecting the most effective processes for waste treatment and disposal. The petitioner's request that "best available technology" be used in removing HLW material from the tank wastes was seen as ignoring costs of disposal, exposures to workers, and environmental impacts.

Some comments disputed the petitioner's claim that the rulemaking proposed in the petition would offer a better process for classification and disposal of the Hanford tank wastes. These commenters did not see any advantage in the proposed process over the process for classification and disposal currently in use. One comment suggested that the Commission's rulemaking requiring disposal of Greater-than-Class C waste in a geologic repository or Commission-approved alternative (53 FR 17710, May 18, 1988) might force DOE to allocate resources to handle the hazards, rather than to waste further time fruitlessly searching for ways to remove more and more activity from one part of the waste. The action proposed by the petitioners was viewed as not increasing the safety of disposal of the waste.

The Commission believes that adherence to the standard of technical and economic practicality generally reflects agreement with these comments.

B. Creation of a Risk-Based Classification System

Several comments, while noting that the rulemaking proposed by the petition would not do so, favored creation of a risk-based system of radioactive waste classification.

The Commission has previously addressed the costs and benefits of creating a new system of radioactive waste classification. Its rationale for not doing so is outlined in the statement of considerations to the proposed part 61 rulemaking on disposal of Greater-than Class C waste (53 FR 17709, May 18, 1988). Further consideration of these issues is beyond the scope of this proposed rulemaking action.

C. NRC Licensing Authority

Some comments focused on the licensing authority of NRC over the Hanford tank wastes. DOE stated that the rulemaking suggested in the petition would involve NRC in regulation of DOE's predisposal waste treatment and processing activities, which would be inconsistent with NRC authority to license specific DOE facilities under the Energy Reorganization Act of 1974. Another commenter stated that the proposed rulemaking was inconsistent with the statutory responsibilities of DOE and NRC. These arguments have already been discussed, and require no further response. It may be emphasized, however, that even if the Commission were found to have jurisdiction over the disposal facilities, it would not regulate either the tanks themselves or the facilities being used to process the wastes in those tanks; and there is reason for concern that implementation of the petitioner's proposal might draw the Commission improperly into regulation of those facilities.

A commenter concluded that DOE was currently in violation of 10 CFR part 30 requirements for a license because various near-surface waste disposal facilities at Hanford are being used for "long-term storage" of high-level radioactive waste. The issue is not pertinent to the subject matter of the petition. However, in any case, the comment does not take into consideration the judicial interpretation of the term in *Natural Resources Defense Council, Inc. v. U.S. Nuclear Regulatory Commission*, 606 F.2d 1261 (D.C. Cir., 1979). The D.C. Circuit Court of Appeals ruled in this case in support of NRC's position that the tanks have not been authorized for use as long-term storage or disposal and are, therefore, not subject to NRC licensing.

D. Public Input

A number of comments stressed the importance of adequate public input into decision making regarding disposal of the Hanford tank wastes. Some called for public hearings on this subject to be held in the Pacific Northwest. One commenter noted that the EIS which was done for Hanford provided the

opportunity for public comment. Another commenter believed that the Commission's rulemaking procedures did not offer the public a better opportunity for input than does the current licensing procedure.

As indicated in the Discussion above the NRC's review of the situation with respect to the double-walled tanks has been carried out publicly from the start. Meetings with DOE have been open, and at least one of the petitioners (the State of Washington) has been provided advance notice and an opportunity to attend. Documents have been placed in the Public Document Room and have been made available for public inspection. It appears to the Commission that the essence of the issue concerns the appropriate standard for evaluating whether certain wastes should be regarded as high-level waste or not. Sufficient factual information is available to carry out these evaluations. Also, the petition for rulemaking has afforded an opportunity for views to be expressed with respect to the appropriateness of the standard.

A decision that NRC lacks licensing jurisdiction does not mean that opportunities for public input will be denied. As DOE undertakes its waste management activities, it will afford opportunities for public participation to the extent required by its own enabling statutes, regulations, and orders.

E. Other Comments

One commenter took exception to the petitioner's claim that the radioactive inventory of the Hanford tank wastes was inadequately known. The commenter believed that the contents of the tanks can be bounded well enough to judge the relative safety of various disposal options.

The Commission considers the available information to be sufficiently bounded to enable it to conclude that DOE's proposed operations (with respect to the material stored in the double-shell tanks) can result in the removal from the Hanford double-shell tanks of as much of the radioactive waste as may be technically and economically practical, and that the applicable regulatory objectives have been satisfied. Once these judgments are made, it is not the NRC's role to judge the relative safety of various disposal options, and we decline to do so.

One comment stated that while the petition was aimed solely at the Hanford tank wastes, its provisions could potentially affect all radioactive wastes from reprocessing, including those at Savannah River, West Valley, and the Idaho National Engineering Laboratory. As the waste management programs at these other sites are in different stages of implementation, the impacts of the provisions would vary from site to site. As indicated above, the Commission is sensitive to this consideration yet believes that the specific case at hand only needs to be addressed at this time.

Some comments urged the Commission not to change the present definition of HLW. The Commission is not changing the present definition.

V. Conclusion

For the reasons presented in this document, the petition for rulemaking is denied.

Dated at Rockville, Maryland this 26th day of February, 1993.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,

Secretary of the Commission.

[FR Doc. 93-4944 Filed 3-3-93; 8:45 am]

BILLING CODE 7590-01-P

DEPARTMENT OF TRANSPORTATION

Federal Aviation Administration

14 CFR Part 39

[Docket No. 93-NM-05-AD]

Airworthiness Directives; Boeing Model 737 Series Airplanes

AGENCY: Federal Aviation Administration, DOT.

ACTION: Notice of proposed rulemaking (NPRM).

SUMMARY: This document proposes the supersedure of an existing airworthiness directive (AD), applicable to certain Boeing Model 737 series airplanes, that currently requires repetitive inspections of the wing main tank float switch electrical conduits for trapped water, and removal of the water, if found. This action would require installation of grease in the interior of the float switch conduits, which would terminate the requirement for repetitive inspections of the conduits. This action would also expand the applicability of the rule. This proposal is prompted by the development of a modification that would preclude the possibility for water to accumulate in the conduits. The actions specified by the proposed AD are intended to prevent fuel leakage from the wing main tanks, which could propagate down the wing leading edge cavity, on to the respective engine tail pipe, and cause an external fire under the wing.

DATES: Comments must be received by April 27, 1993.

ADDRESSES: Submit comments in triplicate to the Federal Aviation Administration (FAA), Transport Airplane Directorate, ANM-103, Attention: Rules Docket No. 93-NM-05-AD, 1601 Lind Avenue, SW., Renton, Washington 98055-4056. Comments may be inspected at this location between 8 a.m. and 3 p.m., Monday through Friday, except Federal holidays.

The service information referenced in the proposed rule may be obtained from Boeing Commercial Airplane Group, P.O. Box 3707, Seattle, Washington 98124-2207. This information may be examined at the FAA, Transport Airplane Directorate, 1601 Lind Avenue, SW., Renton, Washington.

FOR FURTHER INFORMATION CONTACT: Stephen Bray, Aerospace Engineer, Seattle Aircraft Certification Office, Propulsion Branch, ANM-140S, FAA, Transport Airplane Directorate, 1601 Lind Avenue, SW., Renton, Washington 98055-4056; telephone (206) 227-2681; fax (206) 227-1181.

SUPPLEMENTARY INFORMATION:

Comments Invited

Interested persons are invited to participate in the making of the proposed rule by submitting such written data, views, or arguments as they may desire. Communications shall identify the Rules Docket number and be submitted in triplicate to the address specified above. All communications received on or before the closing date for comments, specified above, will be considered before taking action on the proposed rule. The proposals contained in this notice may be changed in light of the comments received.

Comments are specifically invited on the overall regulatory, economic, environmental, and energy aspects of the proposed rule. All comments submitted will be available, both before and after the closing date for comments, in the Rules Docket for examination by interested persons. A report summarizing each FAA-public contact concerned with the substance of this proposal will be filed in the Rules Docket.

Commenters wishing the FAA to acknowledge receipt of their comments submitted in response to this notice must submit a self-addressed, stamped postcard on which the following statement is made: "Comments to Docket Number 93-NM-05-AD." The postcard will be date stamped and returned to the commenter.

Availability of NPRMs

Any person may obtain a copy of this NPRM by submitting a request to the FAA, Transport Airplane Directorate,

APPENDIX B

COST ESTIMATES

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APPENDIX B

COST ESTIMATES

The study used costs taken from existing sources to the extent possible. These sources include prior engineering studies, conceptual designs, and data packages supporting the TWRS Environmental Impact Statement (in preparation). This appendix provides a reference to the sources used for cost estimation. In addition, the study includes allowances for technologies not completely evaluated by previous engineering studies. Some economic assessments are presented for technologies determined not to be technically practical and are provided for information only.

For the technologies with developed cost estimates, the cost presented represent a Total Estimated Cost (TEC) without escalation applied. The estimates have engineering factors applied depending on the types of facility ranging from 20 to 40 percent and a contingency of 30 to 40 percent.

For the technologies costed by allowance, the allowances have a great deal of conservatism built into them The cost to deploy these technologies should be lower than the cost used in this study. These technologies' costs come from similar or more complex systems.

B1.0 BASELINE PRETREATMENT PROCESS

This baseline pretreatment processing differs from the current TWRS baseline process by the absence of cesium removal using ion exchange. Cesium removal by ion exchange is addressed as a radionuclide removal technology option in Section 4.2.2. The baseline pretreatment processes, along with cesium ion exchange, were adopted as the basis for the Tri-Party Agreement (Ecology et al. 1994).

The baseline processing results in an estimated total capital, operating, and geologic disposal cost savings of $9 billion over geologic disposal of all tank wastes (no separation of a LAW fraction). The cost savings is determined by difference of the capital, operating, and geologic disposal costs defined in technical data packages for the TWRS Environmental Impact Statement (EIS) for no separations (Colby 1995) and the Tri-Party Agreement alternatives (Slaathaug 1995). The Tri-Party Agreement alternative data package costs are adjusted for this comparison by subtracting the cesium ion exchange portion of the costs to obtain the costs associated with solids/liquids separations (which includes enhanced sludge washing).

B1.1 CESIUM REMOVAL COSTS

This section reviews the cost basis for the cases carried in the document to remove cesium from the tank waste supernatant. The technologies costed include cesium volatility, single-cycle cesium ion exchange, second-cycle cesium ion exchange, CST in facility, selective CST in tank, and selective in facility single-cycle cesium ion exchange.

B1.2 CESIUM VOLATILIZATION

An allowance of $30 million for capital and $20 million for operations is used for cesium volatilization (Section 3.2.3). The base line facilities contain much of the equipment to support cesium recycle to the melter. To remove the volatilized cesium, it is necessary to modify the equipment design and configuration. The modification include separations equipment for volatile species.

An estimate is not specifically available for this equipment. However far more complex modification of the melter offgas system was estimated by Fluor Daniel Inc. (*Chloride Removal from Vitrification Offgas*, WHC-SD-WM-TI-702, Rev. 0, May 23, 1995, E. J. Slaathaug, pages 19-20) for WHC to estimate chloride removal systems. Table B-1 shows the results of that study.

Table B-1. Chloride and Fluoride Removal System Costs.

Cost element	Cost ($ millions)
Chloride and fluoride purge distillation and grouting equipment	48.4
Facility differential	8.7
Annual operating	8.3

The chloride and fluoride estimate specified Hastelloy and Inconel materials in the equipment to withstand the corrosion which cost more than stainless steel necessary for the cesium removal system.

What can be learned from this comparison is that cesium recovery equipment is required to recycle cesium back to the melter under any scenario. The delta above the recycle capability to route the cesium to HLW storage is conservatively bracketed by the above chloride estimates at $30 million capital and $20 million operating. A preliminary material balance shows no need for the addition of extensive unit operations to support this option.

B1.3 SINGLE-CYCLE ION EXCHANGE

The cost for cesium removal by ion exchange is based on data in the TWRS EIS data package (Slaathaug 1995, Table F-36). The capital cost for cesium ion exchanges is $380 million. The total operational cost for the cesium ion exchange system including research and development, start-up, operation, and Decontamination and Decommissioning (D&D) costs is $602 million.

This design used a transfer aisle facility design as the basis for cost. The development of these costs come from detail material and energy balances, staffing estimates, equipment lists, and facility layouts. The basis for this estimate has evolved over a five-year period.

Table B-2. Cost from Slaathaug 1995, Table F-36.

Cost element	Cost ($ millions)
Research and development	83
Capital (e.g., construction)	380
Labor for start-up, operations, and decontamination and decommissioning	276
Equipment Replacement	57
Materials for start-up, operations, decontamination and decommissioning	186
Total	982

B1.4 SECOND-CYCLE ION EXCHANGE

The costs associated with a second cycle of cesium ion exchange are derived from the difference in costs for separations facilities containing two cycles and one cycle of cesium ion exchange developed in the TWRS Facility Configuration Study (Boomer et al. 1994). The second cycle of cesium ion exchange increased the capital estimate by $275 million (Table 6.3-1, Boomer et al. 1994). The facilities' cost come from a low maintenance facility design. This facility design has a lower estimate than the facility design used for the single-cycle cesium ion exchange. If the transfer aisle design was used, this delta would increase.

The operating cost difference for the second cycle of cesium ion exchange is $47 million (Table 4.1-4, Boomer et al. 1993). The second-cycle cesium ion exchange has increased equipment replacement cost of $15 million and increased D&D cost of $83 million based on the capital cost delta of $275 million. The other costs associated with cesium ion exchange were estimated not to change between the single-cycle and double-cycle ion exchange.

Table B-3. Cost Differences from Boomer et al. 1994.

Cost element	Single-cycle cesium ion exchange cost ($ millions)	Two cycles of cesium ion exchange cost ($ millions)	Cost differences ($ millions)
Capital (e.g. construction)	426	701	275
Materials for operations	96	143	47
Equipment Replacement	45	60	15
Decontamination and decommissioning	194	277	83
Total	761	1,181	420

B1.5 CESIUM SORPTION BY CST TREATMENT OF ALL WASTES

Costs for use of CST in columns for all the tank waste is analogous to single-cycle ion exchange in columns described above. Since CST is not regenerated, the facility costs are lowered because regeneration equipment and floor space are not required. The reduced facility size decreases the labor cost for operation, but the material cost for CST more than off sets this and reduction and raises the operational cost for this alternative above single-cycle cesium ion exchange. Also the use of CST increases the volume of HLW. This increase has an estimated value of $745,000 per canister.

A detailed cost estimate was not developed for CST columns and allowances have been developed. Table B-4 shows the cost for a CST facility compared to the single-cycle ion exchange facility. The total operating cost for pretreatment and the HLW treatment delta is $1,860 million ($1,670 million in HLW costs and $198 million operating) for CST treatment of all waste to remove cesium. The capital and operating total is $1,688 million.

Table B-4. High-Level Waste Cost from Slaathaug 1995, Table F-36.

Cost element	Single-cycle ion exchange cost from the EIS data packet ($ millions)	Crystalline silicotitanate cost allowance ($ millions)	Cost difference ($ millions)
Capital (e.g., construction)	380	200	(180)
Operational cost	602	800	198
High level waste delta cost for an incremental 2,240 canisters	0	1,670	1,670
Total	982	2,670	1,688

The cost used for calculating the incremental cost per HLW canister come from the EIS data packet for the Tri-Party Agreement Alternative (Slaathaug 1995, Table F-36). The HLW cost from the data packet consist of fixed and variable cost. Table B-5 shows the split between the cost types. The fixed cost consisted of research and development costs and capital cost both of which do not factor into the cost of producing an extra canister of glass. The repository cost increases by the amount of $745 thousand per incremental HLW canister as shown below (Slaathaug 1995, Table F-36), based on a variable cost of $5.065 billion for 6,800 canisters.

Table B-5. High-Level Waste Cost from Slaathaug 1995, Table F-36.

Cost element	Fixed cost ($ millions)	Variable cost ($ millions)
Research and development	260	
Capital (e.g., construction)	1,400	
Labor for start-up, operations, and decontamination and decommissioning		639
Equipment Replacement		70
Materials for start-up, operations, decontamination and decommissioning		604
High level waste canisters		239
Transportation		31
Repository fee		3,960
Total	1,600	5,065

B1.6 CST TREATMENT OF SELECTED TANKS

Prior estimates do not provide a direct basis for CST addition to double-shell tanks. The process of adding the CST slurry to a double-shell tank and obtaining the required contact efficiency is simplified by in tank washing and retrieval operations necessary for other reasons. CST addition for selected tanks is held below the level at which titanium impacts the HLW glass loading. The remaining costs include the capital to add CST to the tanks as a slurry and a small operating delta to accommodate the mixing via operation of existing mixer pumps.

The operating cost per incremental HLW canister is approximately $745,000 (Slaathaug 1995, Table F-36). This includes labor, materials, and geologic disposal. The geologic disposal costs are assumed to be $582,000 per canister assuming two repositories (Slaathaug 1995). An allowance of $2 million per tank of treated waste is assumed for pretreatment operating costs including development, chemical purchases, and incremental tank farm operations to perform the batch contacts and decants.

An allowance of $5 million is provided for capital cost requirements associated with addition of CST slurries to the wastes before or during the retrieval step. It is assumed that the retrieval operation will provide adequate contact with the CST solids before the following settle/decant operation. The operating costs associated with addition of the CST slurries are $25 million for operating labor and CST purchase costs. The selective treatment of wastes with CST assumes two CST costs to bound the analysis. The currently quoted price for small quantities is $176/kg of CST powder. An alternate price for CST costs with the large amounts envisioned by this process was assumed to be equal to $1/Ci of ^{137}Cs removed. This assumed large volume price break results in a assumed unit price reduction of 40 to 85 percent.

B1.7 PHASED PROCESSING AND MODULAR FACILITIES

During development of the study the use of selective CST showed a favorable cost as compared to global application of cesium ion exchange. As such, the study revisited the single-cycle cesium ion exchange data to find an analogous application. Table B-6 summarizes three facility cases for application of cesium ion exchange. The cases and specific sources of capital and operating costs are as follows:

1. Full Treatment (treatment of all tanks in a full size facility) is taken from the EIS data package. WHC-SD-WM-EV-104, Rev.0-A *TPA Alternative Engineering Data Package for the TWRS EIS* E. J. Slaathaug, July 1993, Table F-36.

2. The modular treatment concept for a 15 GPM capacity was developed by BNFL as a facility configuration study (FCS) alternative for cesium removal. The costs in the table are for treatment of 12 DST's containing the majority of the cesium (NCAW, CC, DSS/DSSF). WHC-SD-WM-ES-295 Rev. 0, *TWRS Facility Configuration Study*, K. D. Boomer, July 13, 1994, Table 6.2-1.

3. Modular treatment at 5 GPM supporting the privatization RFP was developed at the request from DOE and documented in two letters; the first for capital costs and the second for operating costs. Letter 9551934, May 1, 1995, Privatization Cost Estimate (Business Sensitive), J. S. Garfield to D. L. Veith, Letter 9552911, May 26, 1995, "Estimate of Privatization Operation Cost" J. S. Garfield to L. S. Waldorf.

The costs in the table are for treatment of 12 DST's containing the majority of the cesium (NCAW, CC, DSS/DSSF).

Table B-6 Global and Selective Cesium Ion Exchange Costs.

Facility/estimate basis	Full treatment, 177 tanks[1]	Modular[2]	Modular[3]
Throughput (GPM)	*25 @ 7M*	*15 @ 5M*	*5 @ 5M*
Yrs to Process 12 DSTs	2	5	14
Costs ($ millions):			
Capital Construction	38	170	100
- Maintenance	—	20	20
- Analytical	—	20	20
Operating Labor (LCC)	276	100	276
Replacement Equipment (LCC)	57	15	50
Materials & Supplies			
- Startup	21	20	10
- D&D	109	60	40
- Chemicals & Consumables	56	10	10
R&D	83	83	83
Total	982	493	609
(Totals Used)	(1,000)	(500)	(600)

[1]Slaathaug, E. J., 1993, *TPA Alternative Engineering Data Package for the TWRS EIS*, WHC-SD-WM-EV-104, Rev. 0-A, Westinghouse Hanford Company, Richland, Washington.

[2]Boomer, K. D., et al. 1994, *TWRS Facility Configuration Study*, WHC-SD-WM-ES-295, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

[3]Garfield, J. S., 1995, *Privatization Cost Estimate (Business Sensitive)*, letter 9551934 to D. L. Veith; and Garfield J. S., 1995, *Estimate of Privatization Operation Cost*, letter 9552911 to L. S. Waldorf.

B2.0 STRONTIUM REMOVAL

This section reviews the cost basis for the cases carried in the document to remove strontium from the tank waste supernatant. The technologies costed include solvent extraction, strontium removal by precipitation as phosphate, selective complexant deactivation, and complexant destruction.

B2.1 STRONTIUM REMOVAL BY SOLVENT EXTRACTION

The cost for strontium removal by solvent extraction is based on facility designs presented in the *Tank Waste Technical Options Report* (Boomer et al. 1993). The cost of a solvent extraction system collocated with another processing facility is based on the cost differences between two pretreatment facilities with and without the solvent extraction system. These two facilities are identified as Sludge Wash B (SWB) and Solvent Extraction B (SOLEX B), with and without the solvent extraction, respectively, in the reference report (Boomer et al. 1993). The capital costs for these two facilities are updated for this report. Summary of the updated SWB is found in Appendix C. The difference in costs between SWB and SOLEX B provides a basis for the cost difference of a LAW vitrification facility with and without solvent extraction capability. Solvent extraction is used for strontium recovery with transuranic extraction (TRUEX) processes. The difference in capital costs associated with solvent extraction is $1.22 billion. The difference in operating costs associated with operation of the solvent extraction system is identified by the difference in operating costs for the SWB and SOLEX B facilities of $6.69 billion (Boomer et al. 1993).

The in-facility process for strontium removal by solvent extraction treatment of ALL wastes is expected to remove 3.4 MCi of the 3.5 MCi soluble and insoluble strontium from the liquid fraction. The removal of 3.4 MCi of ^{90}Sr for a total cost of $7.9 billion results in a removal cost of $2,320/Ci.

B2.2 PRECIPITATION OF STRONTIUM PHOSPHATE

Addition of strontium nitrate to the three CC waste tanks would impact the HLW glass volumes with an increase of 75 HLW canisters at a unit delta of $745,000 each as explained in Section B1-3. The allowance for capital costs can conservatively be zero.

Capital:	=	-0-
Operating: 75 HLW Canisters @ 745,000 each	=	$56 million

B2.3 STRONTIUM REMOVAL BY COMPLEXANT DEACTIVATION IN SELECTED WASTES

Complexant deactivation is accomplished by in-tank heat and digest for the three CC waste tanks affected. The capital and operating cost impacts are $150 million and $50 million respectively. The numbers are developed in the following reference.

> Klem, M. J., *Preliminary Engineering Evaluation of Heat and Digests Treatment for In-Tank Removal of Radionuclides From Complexed Hanford Tank Waste*, WHC-SD-SM-TI-719, Rev. 0, Table 2-3.

B2.4 STRONTIUM REMOVAL BY COMPLEXANT DESTRUCTION IN ALL WASTES

The cost for strontium removal by complexant destruction is based on facility designs presented in the *T : Waste Technical Option: Report* (Boomer et al. 1993). The cost of a complexant destruction system collocated with another processing facility is based on the cost differences between two pretreatment facilities with and without the complexant destruction system. These two facilities are identified as Sludge Wash C (SWC) and Sludge Wash B (SWB), with and without the complexant destruction, respectively, in the reference report (Boomer et al. 1993).

The capital costs for these two facilities are updated for this report. Summaries of the updated SWC and SWB cost estimates are found in Section B7.0 The difference in costs between SWC and SWB provides a basis for the cost difference of a LAW vitrification facility with and without complexant destruction capability. The difference in capital costs associated with strontium removal by complexant destruction is $381 million. The difference in operating costs associated with operation of the complexant destruction system is identified by the difference in operating costs for the SWC and SWD facilities of $330 million for training, operations, essential materials, and decontamination and decommissioning (Boomer et al. 1993, Table G16-15). Table B-7 shows the costs for organic destruction.

Table B-7. Costs for Organic Destruction.

Cost estimate	Capital cost ($ millions)	Other cost ($ millions)	Total cost ($ millions)
Sludge Wash C	1,913	2,670	4,583
Sludge Wash B	1,532	2,340	3,872
Delta	381	330	711

B3.0 TRANSURANIC REMOVAL

This section reviews the cost basis for the cases carried in the document to remove TRU from the tank waste supernatant.

B3.1 HYDROXIDE PRECIPITATION OF TRU FROM SELECTED WASTES

This option entails addition of 19M NaOH and .01M ferric ion to the three CC waste tanks to achieve a 3M hydroxide concentration to precipitate the TRU. The allowance for capital impacts from adding chemicals to the CC waste tanks is considered negligible. Features to add chemicals are presently available and mixing is accomplished by the mixer pumps required for retrieval.

The total operating costs for adjustment of the hydroxide concentration and addition of 0.01*M* iron in the DSTs several weeks or months before retrieval and solids/liquid separation is $6 million to $18 million as summarized in Table B-8:

Table B-8. Hydroxide Precipitation Costs.

Cost element	Cost ($ millions)
Sodium hydroxide	0.6
Additional low-activity waste vaults (for 0.14 Vaults)	1.4
Iron impact on high-level waste (additional 5 to 21 canisters)	3.7 to 16
Total operating	6 to 18

A LAW cost increase of $2 million is based on addition of 19*M* NaOH to the three tanks to a 3*M* NaOH final concentration which increases the total tank waste sodium by 250 MT or 0.14 equivalent vaults. The disposal cost per vault of solidified LAW is approximately $10 million (Slaathaug 1995, Table F-36). This includes containers, sulfur matrix, and vault costs. The incremental Na usage results in $1.4 million increase in LAW disposal costs. The addition of NaOH to the tanks is estimated to cost $0.6 million.

The addition of the 0.01*M* iron results in an additional 5 to 21 canisters of HLW glass. The canisters are assumed to contain 1.26 m^3 of glass and a maximum Fe_2O_3 content of 12 wt% (Orme 1994). The minimum HLW canister impact is based on 50 wt% Fe_2O_3 if the HLW facility is processing glass at less than the 12 wt% Fe_2O_3 limit and the maximum HLW canister impact is based on the 12 wt% Fe_2O_3 limit. With an incremental HLW canister cost

of $745,000 per HLW canister (Slaathaug 1995), this results in an additional operating cost of approximately $3.7 to $16 million.

B3.2 TRANSURANIC REMOVAL FROM SELECTED WASTES BY FERRIC HYDROXIDE PRECIPITATION

The addition of 0.1*M* $Fe(NO_3)_3$ and 3.0*M* NaOH for precipitation of TRU to the three CC tanks results in a maximum additional 50 to 210 canisters of HLW glass. The canisters are assumed to contain 1.26 m^3 of glass and a maximum Fe_2O_3 content of 12 wt% (Orme 1994). The minimum HLW canister impact is based on 50 wt% Fe_2O_3 if the HLW facility is processing glass at less than the 12 wt% Fe_2O_3 limit and the maximum HLW canister impact is based on the 12 wt% Fe_2O_3 limit. With an incremental HLW canister cost of $745,000 per HLW canister (Slaathaug 1995), this results in an additional operating cost of approximately $37 to 160 million. The incremental LAW costs are $2 million as developed in Section C3-1. Totals are summarized as follows:

NaOH chemical Costs	$0.6 million
LAW Vault incremental cost (0.14 Vaults)	$1.4 million
Iron impact on HLW (50-210 Canisters)	$37-160 million
Total operating Cost	$39-162 million

B3.3 TRANSURANIC REMOVAL BY COMPLEXANT DEACTIVATION IN SELECTED WASTES

Complexant destruction is accomplished with in tank heat and digest for the three CC waste tanks affected. The capital and operating cost impacts are $150 million and $50 million respectively. The numbers are substantiated in the following reference.

Klem, M. J., *Preliminary Engineering Evaluation of Heat and Digests Treatment for In-Tank Removal of Radionuclides From Complexed Hanford Tank Waste*, WHC-SD-SM-TI-719, Rev. 0, Table 2-3.

B3.4 TRANSURANIC REMOVAL BY COMPLEXANT DESTRUCTION IN ALL WASTES

The explanation for the cost basis for this option is identical to complexant destruction for strontium removal discussed in Section B2.3.

B3.5 TRANSURANIC REMOVAL BY SOLVENT EXTRACTION

The explanation of the cost basis for this option is identical to strontium solvent extraction discussed in Section B2.1.

B4.0 TECHNETIUM-99 REMOVAL

This section reviews the cost basis for the cases carried in the document to remove technetium from the tank waste supernatant. The technologies include technetium volatilization, mixed bed technetium ion exchange, and technetium anion exchange.

B4.1 TECHNETIUM VOLATILIZATION

The explanation for the cost basis for this option is identical to cesium volatilization discussed in Section B1.1.

B4.2 MIXED BED TECHNETIUM AND CESIUM ION EXCHANGE

The mixed bed ion exchange removal system is assumed to remove 99 percent of the soluble ^{99}Tc, or 0.0216 MCi of the 0.0228 MCi soluble ^{99}Tc. No breakthrough data are available to perform specific equipment sizing or material balance calculations. The potential costs of the mixed bed system assume that the bed volume of the combined resins and the elution volumes are increased 25 percent. It is assumed that the capital and operating costs are increased 10 and 20 percent, respectively, over the costs of a cesium only ion exchange system {see Section C1.2 (Slaathaug 1995, Table F-36)}. The capital and operating costs of the cesium only ion exchange system are $380 and $602 million, respectively. The incremental capital and operating cost increase for the mixed bed technetium recovery are $38 and $120 million, respectively. The total incremental cost of 158 million results in a cost of $7,200/Ci of removed ^{99}Tc.

B4.3 TECHNETIUM ANION EXCHANGE

The cost for technetium removal by ion exchange is based on facility and equipments designs presented in the *Tank Waste Technical Options Report* (Boomer et al. 1993). The cost of a technetium ion exchange system collocated with another processing facility is based on the cost differences between two pretreatment facilities with and without the technetium ion exchange process. These two facilities are identified as Sludge Wash D (SWD) and Sludge Wash B (SWB), with and without the technetium ion exchange, respectively, in the reference report (Boomer et al. 1993).

Both facilities include sludge washing, solids/liquids separation, and cesium ion exchange. The capital costs for these two facilities are updated for this report. Summaries of the updated SWD and SWB cost estimates are found in Section C7.0. The difference in costs between SWD and SWB provides a basis for the cost difference of a LAW treatment or vitrification facility with and without technetium ion exchange capability. The difference in capital costs associated with technetium ion exchange processing is $260 million. The difference in operating costs associated with operation of the technetium ion exchange processing is identified by the difference in operating costs for the SWD and SWB facilities

of $400 million for training, operations, associated materials, and decontamination and decommissioning (Boomer et al. 1995, Table G16-15). Table B-9 provides the cost for technetium anion exchange.

The technetium ion exchange process recovers 0.0226 MCi of ^{99}Tc for a total cost of $660 million. This results in a cost of $29,000/Ci ^{99}Tc.

Table B-9. Costs for Technetium Anion Exchange.

Cost estimate	Capital cost ($ millions)	Other cost ($ millions)	Total cost ($ millions)
Sludge Wash C	1,532	2,340	3,872
Sludge Wash B	1,272	1,940	3,212
Delta	260	400	660

B5.0 SELENIUM REMOVAL

Selenium removal is accomplished by volatilization in the LAW melter. The cost is identical to cesium removal discussed in Section B1.1. Selenium removal has no incremental cost above cesium and technetium. Selenium volatilization is not technically practical. The economics provided below are for information only.

The melter volatility system is assumed to remove 80 percent of the feed Se (DF of 5). The melter volatility system thus removes 0.00082 MCi of the 0.00103 MCi ^{79}Se. Some volatilization also occurs for ^{137}Cs and ^{99}Tc. For ^{137}Cs, it assumed 50 percent of the feed cesium volatilizes (0.5 MCi) and for ^{99}Tc, 0.0205 MCi of the 0.0228 MCi soluble ^{99}Tc volatilizes. The melter and offgas system that provides radionuclide removal costs approximately $50 million. This results in a cost of $1.00 per Ci.

B6.0 CARBON-14 REMOVAL

The chemistry of carbon results in ^{14}C being distributed in supernatants and solids of all tanks. The organic carbon and carbonate content of the wastes will be converted to carbon dioxide, CO_2, as a result of the LAW vitrification process. In the vitrification process a total of 6,500,000 kg of CO_2 will be released in the offgas. The 6,500,000 kg of CO_2 can be absorbed in slaked lime as 15,000,000 kg of $CaCO_3$.

The recovery, packaging and geologic disposal of 15,000 Mg of $CaCO_3$ in approximately 6,000 1.26-m^3 canisters would cost approximately $5 billion. A cost of $5 billion to mitigate the environmental release of 0.0053 MCi ^{14}C results in a cost of $1,000 000/Ci and is not economically practical.

B7.0 URANIUM REMOVAL

This option is bounded by the Solvent Extraction B option discussed in the *Tank Waste Technical Options Report* (Boomer et al. 1993, Tables G16-14 and G16-15). These costs are summarized below.

Capital:	$2,490 million
Operating:	$8,630 million.

B8.0 COST TABLES FOR SLUDGE WASH FACILITIES

The following tables provide cost estimates for the Sludge Wash B (SWB), Sludge Wash C (SWC), and Sludge Wash D (SWD) facilities defined in the *Tank Waste Technical Options Report*, WHC-EP-0616 (Boomer et al. 1993). The cost estimates in the reference were updated for this document to assure a consistent basis for the incremental costs of single process technology options. The cost estimates have been escalated to 1995 dollars.

APPENDIX C

SOURCE OF TANK WASTE AT THE HANFORD SITE

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APPENDIX C

SOURCE OF TANK WASTE AT THE HANFORD SITE

The Manhattan Project established the Hanford Site in 1943 to produce plutonium for nuclear weapons in support of World War II. Plutonium production continued until January 1987 when the last production reactor ceased operation at the Hanford Site. Eight production reactors, one dual-purpose reactor (N Reactor), and five reprocessing facilities operated at the Hanford Site to support that mission. These operations created a large quantity of radioactive wastes, much of which continues to be stored in 149 SSTs and 28 DSTs.

Figure C-1 shows the cesium, strontium, technetium, and transuranic curie balance for the Hanford Site facilities, decayed to December 31, 1999. The figure depicts the route of cesium and strontium bearing waste into and out of the tank complex.

The radionuclide inventory of 178 MCi (December 31, 1999, decay date) for waste entering the tanks shown in Figure C-1 is the sum of inventories in the tanks and known removals from the tanks described in Sections C-2 and C-3.

C1.0 WASTE ENTERING THE TANKS

The waste entering the tank came primarily from the reprocessing of spent nuclear fuel. The following sections provide a description of the wastes and other wastes discharged to the tanks.

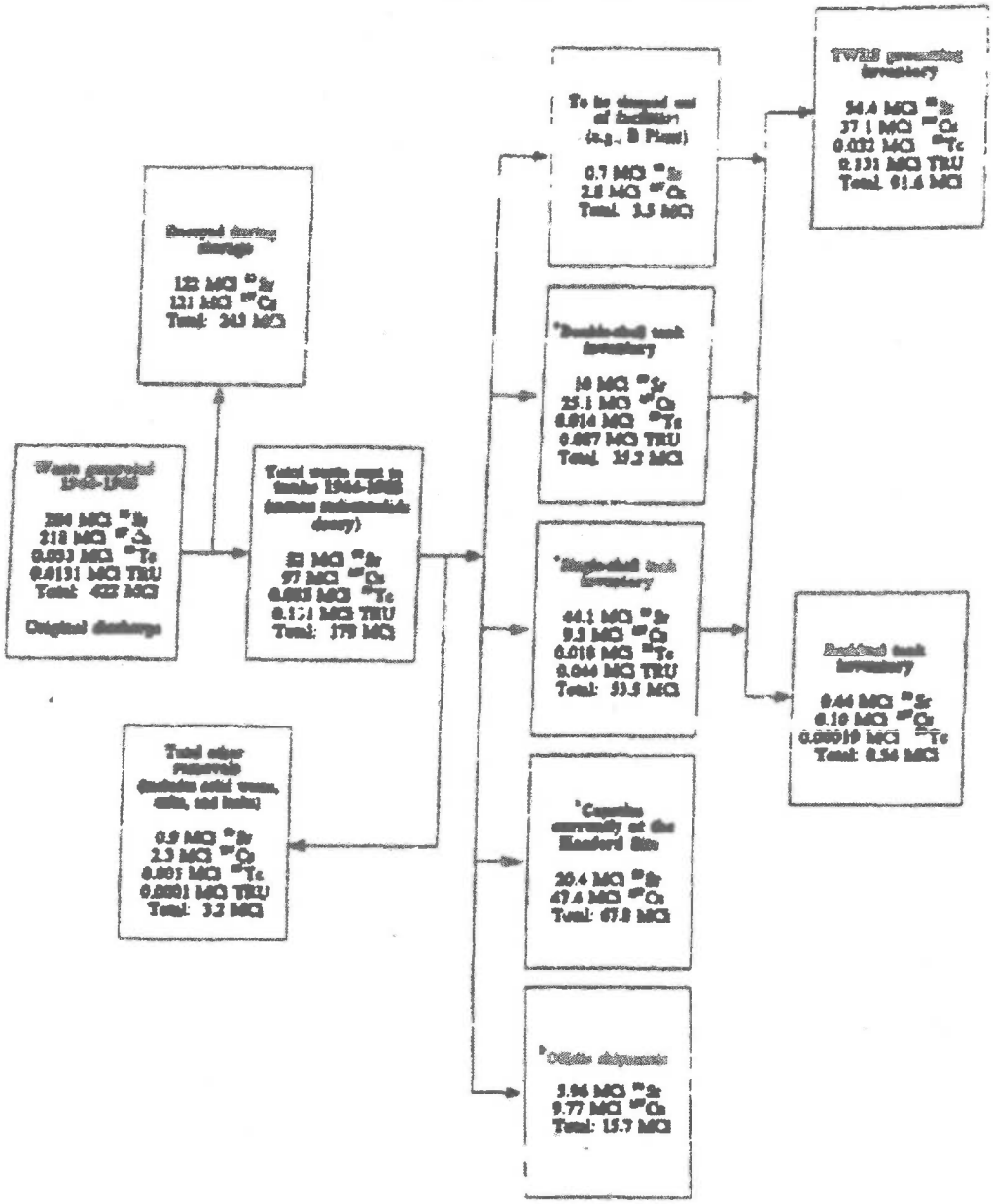
C1.1 FUEL REPROCESSING WASTE

The Hanford Site used three different fuel reprocessing processes on irradiated fuel discharged from the reactors: Bismuth Phosphate (BiPO4), REDOX, and PUREX processes. This section provides a brief description of the processes and resulting reprocessing wastes discharged to the tanks.

C1.1.1 Bismuth Phosphate Waste

T Plant and B Plant discharged waste to the tanks from the $BiPO_4$ process. T Plant operated form 1944 through 1956. B Plant operated from 1946 through 1952. This carrier precipitation process discharged four wastes to the tank farms: metal, first cycle, second cycle, and 224 wastes. This process recovered plutonium from fuel containing about four percent of all the activity discharged to the tanks.

Figure C-1. Estimated Hanford Site Tank Waste Radionuclide Inventory.[c,d]



[a]Curie values are based on the Integrated Data Base Report-1994, Rev. 11, Table 2.11 decayed to December 31, 1999.

[b]The "offsite shipments" inventory is not expected to return to the Hanford Site for treatment.

[c]Decay products are not listed. Some radionuclides, such as ^{137}Cs and ^{90}Sr, have daughters with relatively short half-lives. However, they are present in concentrations associated with the normal decay chains of the radionuclides.

[d]Inventories of key radionuclides (i.e. ^{3}H, ^{129}I, ^{14}C, ^{79}Se, uranium isotopes and ^{126}Sn) are not shown on the material balance. These radionuclides have small inventories which do not significantly affect the total curies in the material balance.

The process generated "metal waste" from the initial carrier precipitation step. This waste contained all of the uranium, about 90 percent of the fission product activity, and one percent of the plutonium from the original feed to the process. The process neutralized this acidic waste to a pH of 7 with sodium hydroxide (NaOH) and treated it with an excess of sodium carbonate before discharging waste to the tanks.

The process generated the first- and second-cycle waste from the second and third carrier precipitation steps. These wastes contained about 10 percent of the fission product activity and two percent of the plutonium from the original feed to the process. The process made these acidic waste alkaline by the addition of NaOH before discharging it to the tanks.

The process generated the 224 waste during the final purification of the plutonium product. This waste contained almost none of the fission product activity (< 0.001 percent) and about 0.5 percent of the plutonium from the original feed to the process. The process neutralized and concentrated this waste before discharge to the tanks.

C1.1.2 REDOX Waste

S Plant discharged waste from the REDOX process from 1952 through 1967. The waste from this process consisted of HLW components from a methlyisobutylketone solvent extraction process. The REDOX Plant processed about 20 percent of the irradiated fuel discharged from the reactors.

These waste contained essentially all of the fission products in the original feed and small quantities (less than 0.5 percent) of the original plutonium. They contained large amounts of aluminum nitrate used as a salting agent in the solvent extraction process and minor amounts of iron and chromium.

C1.1.3 PUREX Waste

The PUREX Plant discharged waste from reprocessing of irradiated fuel to the tank farms from 1956 through 1967 and again from 1983 through 1988. The PUREX process used tributyl phosphate (TBP) diluted with kerosene as the extractant. The HLW generated by the PUREX process contained essentially all of the fission products in the original feed plus small amounts of plutonium and uranium also in the waste (< 0.2 percent). The PUREX Plant reprocessed about 76 percent of the irradiated fuel discharged from the reactors.

Before discharging this waste to the tanks, process steps made the PUREX HLW (principally nitric acid) alkaline by the addition of NaOH. This method of direct storage was used from 1956 through 1967. Beginning in 1968 through 1972, the PUREX Plant transferred the Current Acid Waste (CAW) directly to the fractionation process. As discussed below, the fractionation process removed the cesium and strontium from the waste and returned low heat waste to the tanks. With restart of the PUREX Plant from 1983 through 1988, the neutralized waste was routed to DSTs designed to hold aging waste.

C1.2 URANIUM RECOVERY WASTE

U Plant discharged waste from the uranium recovery process from 1952 through 1957. This process used acid to dissolve uranium-bearing $BiPO_4$ "metal wastes" sluiced from the storage tanks. A solvent extraction process using TBP as the extractant recovered the uranium from the acid feed. After uranium recovery, the process re-neutralized the acidified wastes, treated the waste to precipitate cesium and strontium as $CsNiFe(CN)_4$ and $Sr_3(PO_4)_2$, and discharged the resulting slurries to SSTs to allow the solids to settle

C1.3 WASTE FRACTIONATION

In the 1960's, B Plant was modified to recover cesium from the tank waste supernatants and strontium from tank waste sludges. This processing removed decay heat from the waste to allow concentration and in-tank solidification. The process neutralized and returned low heat wastes to the tanks for subsequent concentration and solidification. Wastes were discharged from this process to the tanks from 1968 through 1984.

The use of chelating agents in the fractionation of strontium caused generation of a unique waste type known as CC waste. The organic compounds (e.g., citric acid, ethylenediaminetetraacetic acid) have caused both safety and process concerns in the tank farms. The degradation of the organic compounds has led to elevated hydrogen levels in the tank space above the waste. These same organic compounds solubilize both strontium and transuranic radionuclides which leads to increased concentrations in the supernatants.

C1.4 PLUTONIUM FINISHING PLANT WASTE

The PFP discharged waste from plutonium processing to the tanks. In recent years, the PFP discharges have gone to tank 241-SY-102 in the 200 West Area.

C1.5 LABORATORY WASTE

The waste from laboratories supporting processing at the Hanford Site discharge to the tanks.

C1.6 DECONTAMINATION WASTE

The tanks have received decontamination waste from T Plant and by rail car shipments from the 100 Areas. T Plant decontamination waste comes from radionuclides removed from equipment before repair and reuse in the plants (e.g., centrifuges). The 100 Area waste came from decontamination of reactor systems such as the N Reactor primary coolant system.

C2.0 WASTE EXITING THE TANKS

Waste has exited the tanks by waste fractionation and by-product recovery, planned and unplanned releases to the soil column, and as contamination on failed equipment. The following sections provide a description of the waste removals from the tanks.

C2.1 WASTE FRACTIONATION AND BY-PRODUCT RECOVERY

In the past, the waste has undergone extensive processing for by-product recovery as radionuclides packaged in capsules for interim storage and as direct shipment of recovered radionuclides in casks to other DOE sites.

C2.1.1 Cesium and Strontium Capsules

Beginning in 1967 cesium and strontium were extracted from the tank waste and purified in B Plant. The fractionated cesium and strontium were encapsulated in the Waste Encapsulation and Storage Facility (WESF). In addition to waste from the tanks, B Plant also received the CAW stream from the PUREX Plant for fractionation and subsequent encapsulation. The DOE stores the capsules in water pools in WESF. The DOE has sent many of the capsules offsite for use in laboratory testing and commercial irradiation, most have been returned.

Table C-1 shows the capsule inventories identified in the July 1995 capsule location quarterly reports (Bender 1995a, b). The quarterly reports give the total number of cesium and strontium capsules produced, the number of capsules located onsite and offsite, and the number of capsules that have been destroyed. In addition, through the effort of the B Plant Capsule Return Program, 95 additional cesium capsules are expected to return from offsite before December 31, 1999. The current radionuclide inventories given in the quarterly reports were decayed to December 31, 1999, for consistency in this report. Table C-1 presents the capsule inventory decayed to December 31, 1999.

Table C-1. Capsule Inventory, MCi (December 31, 1999, Decay Date).

Capsules	Original (MCi)	Other uses/ disposed by others (MCi)	Projected inventory[1] (MCi)
^{137}Cs	54.8	7.29	47.5
^{90}Sr	23.0	2.06	20.9
Total	77.8	9.35	68.4
Total including daughters (^{137}Ba, ^{90}Y)	152.8	18.4	134.4

[1]Projected inventory values are generally consistent with the Integrated Data Base Report-1994 (i.e., 47.4 MCi ^{137}Cs and 20.4 MCi ^{90}Sr).

C2.1.2 Hanford Atomic Products Operations Cask Shipments

The Hanford Site began shipping by-product material to the Oak Ridge National Laboratory in 1961 using Hanford Atomic Products Operations (HAPO) casks. The waste shipped in these casks came primarily from PUREX waste and was in the form of ^{137}Cs loaded or an elutable zeolite or ^{90}Sr as a $SrCO_3$ precipitate. The total amount of radionuclides removed from the Hanford Site in HAPO cask shipments is 3.90 MCi of ^{90}Sr and 2.48 MCi of ^{137}Cs (decay date December 31, 1999), for a total of 6.38 MCi shipped offsite.

C2.2 PLANNED AND UNPLANNED RELEASES TO THE SOIL COLUMN

The soil column at the Hanford Site contains radionuclides from the waste reprocessing operations. These radionuclides came both from planned and unplanned releases.

C2.2.1 Planned Releases

In the past, the Hanford Site has used the soil column in the management of radioactive waste. Some of the SST waste was intentionally discharged to the ground via pumping and cascade overflow from 1946 through 1966. The waste discharged to the ground were those that had a low radionuclide concentration which did not require tank storage, but a radionuclide concentration too high to allow discharge to surface ponds. These discharges included: $BiPO_4$ waste, uranium recovery process waste, laboratory wastes, and equipment decontamination wastes which were routed through SSTs before discharge to the ground. Table C-2 shows the waste discharged from the tanks to the soil column (Waite 1992).

Table C-2. Estimated Discharges from Tanks to the Soil Column (December 31, 1999, Decay Date).

Radionuclide	MCi
^{137}Cs	0.0131
^{90}Sr	0.0108
^{99}Tc	0.0009
Transuranics	0.0007

Also, non-SST waste such as process condensate waste streams from evaporator concentration have been discharged intentionally to the ground from both waste concentration activities and from the process plants. These sources are not significant contributors of ^{90}Sr or ^{137}Cs to soil column discharges.

C2.2.2 Unplanned Releases

In addition, unintentional discharge of SST waste has resulted from leaks in tanks and transfer lines and miscellaneous spills. Table C-3, Contaminated Soil Inventory, summarizes the unintentional discharges of SST wastes. The unplanned releases of ^{99}Tc, ^{129}I, and nitrate to the soil column was described in the *Tank Waste Technical Options Report* (Table M2-12, Boomer et al. 1993). The unplanned release of ^{137}Cs is reported as 0.290 to 1.090 MCi (Hanlon 1995). The contaminated soil inventory for the insoluble ^{90}Sr and TRU radionuclides is assumed to be zero for this study.

Table C-3. Estimated Unplanned Releases to the Soil Column (December 31, 1999, Decay Date).

Radionuclide	MCi
^{90}Sr	0.0
^{99}Tc	0.00014
^{129}I	0.0000004
^{137}Cs	0.290 to 1.090
Transuranics	0.0

C2.3 SOLID WASTE BURIAL

Other deductions in the tank waste inventory in the form of solid waste burial of failed equipment from reprocessing plants, B Plant, and WESF processing are shown in Table C-4. The majority of activity was in failed equipment from B Plant and the encapsulation process. The total solid waste burial is provided by the Solid Waste Inventory and Tracking System (SWITS) database (Anderson and Hagel 1995).

Table C-4. Solid Waste Inventory Summary.

Radionuclide	Total December 1999 Decay MCi
^{90}Sr	0.90
^{99}Tc	-0-
^{137}Cs	1.18

Radioactively contaminated solid waste has been disposed by burial in the ground since the Hanford Site began operation in 1944. Radioactive waste burials were stopped in the 300 Area in 1972 and in the 100 Areas in 1973. Since 1970, waste suspected of containing transuranic nuclides was segregated and retrievably stored. Thus, the data regarding solid waste burial are in three categories as listed in Table C-4.

Transuranic waste in the burial grounds is not included in the total inventory values because the material comes from production, did not pass through the tanks, and plays no role in the mass balance for tank waste.

C3.0 TANK WASTE INVENTORY

The tank waste inventory represents the amount of waste and contained radionuclides that will be ultimately processed by pretreatment and LAW and HLW vitrification. The TWRS processing inventory consists of the current tank inventory and future additions less the tank heel following waste retrieval. The processing inventory is discussed in the following sections.

C3.1 CURRENT TANK INVENTORY

Table C-5 contains the tank waste inventory for selected radionuclides decayed to December 31, 1999 (DOE 1994). The inventories presented here use the data proposed for the 1995 Integrated Data Base report (Shelton 1995), which provides the soluble and insoluble fractions for the tank waste.

Table C-5. Tank Inventory (December 31, 1999, Decay Date).

Radionuclide	Tank waste inventory, MCi		
	Soluble	Insoluble	Total
^{90}Sr	1.89	52.2	54.1
^{137}Cs	31.4	3.01	34.4
TRU	0.00961	0.121	0.131
^{99}Tc	0.0228	0.00930	0.0321
Total	33.3	55.3	88.6
Tank total including daughters (^{137}Ba, ^{90}Y)	65.0	110.0	175.6

C3.2 FUTURE TANK ADDITIONS

The Hanford Site no longer has a production mission and no reprocessing waste will be discharged to the tanks. Facility deactivation provides the primary source of future tank waste. The future tank additions are shown in Table C-6. The majority of this waste will come from B Plant. The flushing of approximately 3 MCi of cesium and strontium from B Plant to the tanks is based on *B Plant Transition Engineering Radioactive Inventory and Material Status (RIMS)* (Gehrke 1995). For mixed inventories reported by Gehrke, the estimated $^{90}Sr/^{137}Cs$ inventory was assumed to be a 50/50 split. The maximum B Plant inventory, reported in a range, is used in this study.

Table C-6. Future Tank Additions.

B Plant Inventory	MCi, December 31, 1999, Decay Date	
	^{90}Sr	^{137}Cs
Radioactive process liquid	0.0146	0.0147
Radioactive filter	0.101	0.587
Residual inactive process and facility	0.0925 to 0.589	0.336 to 2.168
Total	0.2 to 0.7	0.9 to 2.8

C3.3 RESIDUAL TANK INVENTORY

The residual amount of tank inventory contained in the tank heels at completion of the retrieval operation has not yet been determined and is subject to a separate NRC determination. For purposes of this study, the residual tank inventory is assumed to be 0.05 percent of the DST inventory (DOE 1987) and 1 percent of the SST inventory (Ecology 1994). The residual SSTs inventory are based on the Integrated Database Report 1995 inventories. For the DSTs, the inventory is based on the Integrated Database Report 1995 inventories along with the future tank additions (Table C-6). Table C-7 shows the estimated residual tank inventory.

Table C-7. Estimated Residual Tank Inventory.

Tank Waste	MCi, December 31, 1999, Decay Date			
	^{90}Sr	^{137}Cs	^{99}Tc	Transuranics
Current double-shell tank (DST) Inventory	10	25.1	0.014	0.087
Future DST Additions	0.7	2.8	0	0
Estimated Future DST Inventory	10.7	27.9	0.014	0.087
Residual DST Inventory	0.005	0.012	7.0E-06	4.3E-05
Single-shell tank (SST) Inventory	44.1	9.3	0.018	0.044
Residual SST Inventory	0.44	0.093	1.8E-04	4.4E-04
Estimated Residual Tank Inventory	**0.44**	**0.10**	**1.9E-04**	**4.8E-04**

C3.4 TANK WASTE REMEDIATION SYSTEM PROCESSING INVENTORY

Table C-8 presents the total inventory of radionuclides that will be processed by pretreatment. The total inventory includes the current tank inventory (Table C-5, Section C3.1) plus the radionuclide inventory to be flushed from B Plant (Table C-6, Section C3.2) less the residual tank inventory (Table C-7, Section C3.3). The ^{90}Sr additions and deletions are in the insoluble fraction. The ^{137}Cs additions and deletions are in the soluble fraction.

The inventories of ^{3}H, ^{14}C, and ^{129}I are given in Colby (1994). The inventory for ^{79}Se is given in Mann et al. (1995).

Table C-8. Estimated Tank Waste Remediation System Processing Inventory (December 31, 1999, Decay Date).

Radionuclides	Tank Waste Remediation System Processing Inventory, MCi		
	Soluble	Insoluble	Total
^{90}Sr	1.89	52.5	54.4
^{137}Cs	34.1	3.01	37.1
Transuranics	0.00961	0.121	0.131
^{99}Tc	0.0228	0.0093	0.0321
^{3}H	0.01	-	0.01
^{14}C	0.0053	-	0.0053
^{79}Se	0.00103	-	0.00103
^{129}I	0.000051	-	0.000051
Total	36.0	55.6	91.6
Tank total including daughters (^{90}Y, ^{137}Ba)	71.4	110.0	181.4

C4.0 REFERENCES

Anderson, J. D., and D. L. Hagel, 1995, *Summary of Radioactive Solid Waste Received in the 200 Areas During Calendar Year 1994*, WHC-EP-0125-7, Westinghouse Hanford Company, Richland, Washington.

Bender, K. P., 1995a, *Quarterly Report: Location of WESF Cesium Capsules*, Internal Memo 16800-95-KPB-003, Westinghouse Hanford Company, Richland, Washington.

Bender K. P., 1995b, *Quarterly Report: Location of WESF Strontium Capsules*, Internal Memo 16800-95-KPB-004, Westinghouse Hanford Company, Richland, Washington.

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Shelton, L. W., 1995, *Radionuclide Inventory for Single- and Double-Shell Tanks*, Internal Memo 71320-95-002, Westinghouse Hanford Company, Richland, Washington.

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APPENDIX D

DISPOSAL PERFORMANCE REQUIREMENTS COMPARISON

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Table D-1. Comparison Between 10 CFR 61 and DOE Order 5820.2A, Chapter III.

NRC: 10 CFR 61, Subpart C, Performance Objectives	DOE Order 5820.2A, Chapter III	*Hanford Low-Level Tank Waste Interim Performance Assessment* (IPA)
61.40 General Requirements "Land disposal facilities must be sited, designed, operated, closed, and controlled so that reasonable assurance exists that exposures to humans are within the limits established in the performance objectives 61.41 through 61.44."	3b(1) Performance Assessment "Field Organizations with disposal sites shall prepare and maintain a site specific radiological performance assessment for the disposal of waste for the purpose of demonstrating compliance with the performance objectives stated in paragraph 3a." 3b(7) Disposal Site Selection (a) "Disposal site selection criteria (based on planned waste confinment technology) shall be developed for establishing new low-level waste disposal sites." (b)-(e) give specific requirements	The document, WHC-EP-0884, is undergoing internal review. External review is scheduled for FY 97. This document uses existing data to provide an indication early in project life on the technical feasibility of disposing such waste at Hanford. A performance assessment as required by DOE Order 5820.2A is scheduled for submittal in December 2002. Performance objectives are documented in *Performance Objectives of the Tank Waste Remediation System Low-Level Waste Disposal Program*, WHC-EP-0826, Revision 0. The performance objectives were sent to members of the Hanford Advisory Board. The comments received did not require any change to the performance objectives.

Table D-1. Comparison Between 10 CFR 61 and DOE Order 5820.2A, Chapter III.

61.41 Protection of the general population from release of radioactivity "Concentrations of radioactive material which may be released to the general environment in ground water, surface water, air, soil, plants, or animals must not result in an annual dose exceeding the equivalent of 25 millirems to the whole body, 75 millirems to the thyroid, and 25 millirems to any other organ of any member of the public. Reasonable effort should be made to maintain releases of radioactivity in effluents to the general environment as low as is reasonable achievable."	3a(2) Performance Objectives "Assure that external exposure to the waste and concentrations of radioactive material which may be released into surface water, ground water, soil, plants and animals results in an effective dose equivalent that does not exceed 25 mrem/yr to any member of the public. ... Reasonable effort should be made to maintain releases of radioactivity in effluents to the general environment as low as reasonably achievable."	The scenario for protection of the general population used is that water is drawn from a well 100 meters downgradient from the disposal facility and used for a small farm. The compliance time is 10,000 years but calculations are carried out to a longer period of time. The scenario for ALARA is based on the integrated dose being less than 500 person-rem. General use of water from the Columbia River by a population of 5 million is assumed. As proposed by DNFSB Recommendation 94-2, the effect of all Hanford sources on the point of compliance is also calculated. The performance objective is 100 mrem in a year.

Table D-1. Comparison Between 10 CFR 61 and DOE Order 5820.2A, Chapter III.

61.42 Protection of individuals from inadvertent intrusion "Design, operation, and closure of the land disposal facility must ensure protection of any individual inadvertently intruding into the disposal site and occupying the site or contacting the waste at any time after active institutional controls over the disposal site are removed."	3a(3) Performance Objectives "Assure that the committed effective dose equivalents received by individuals who inadvertently may intrude into the facility after the loss of active institutional control (100 years) will not exceed 100 mrem/yr for continuous exposure or 500 mrem for a single acute exposure."	The IPA follows the example of the Hanford grout performance assessment (WHC-SD-EE-004, Rev. 1) which uses a 500 year time of compliance with passive controls and barriers are used. This is also consistent with 10 CFR 61.52a(2) requiring inadvertent protection for 500 years. The scenario for the acute exposure is a person drilling a well through the waste and being exposed to the drill cuttings. The scenario for the continuous exposure is for a homesteader to disperse the drill cuttings throughout a garden.

Table D-1. Comparison Between 10 CFR 61 and DOE Order 5820.2A, Chapter III.

61.43 Protection of individuals during operation "Operations at the land disposal facility must be conducted in compliance with the standards for radiation protection set out in part 20 of this chapter, except for releases of radioactivity in effluents which shall be governed by 61.41 of this part. Every reasonable effort shall be made to maintain radiation exposures as low as is reasonably achievable."	3i(9a) Disposal Facility Operations "Field organizations shall develop and implement operating procedures for low-level waste disposal facilities that protect the environment, health and safety of the public and facility personnel ..."	This requirement is not documented in a performance assessment. Rather other documents (e.g. Safety Analysis Reports) are used.

Table D-1. Comparison Between 10 CFR 61 and DOE Order 5820.2A, Chapter III.

61.44 Stability of the disposal site after closure "The disposal facility must be sited, designed, used, operated, and closed to achieve long-term stability of the disposal site and to eliminate to the extent practicable the need for on-going active maintenance of the disposal site following closure so that only surveillance, monitoring, and minor custodial care are required.	3a(8a) Disposal Facility and Disposal Site Design "Design criteria shall be established prior to the selection of new disposal facilities, new disposal sites, or both. These design criteria shall be based on analyses of physiographic, environmental, and hydrogeological data to assure that the policy and requirements of this Order can be met. ..." 3a(9a) Disposal Facility Operations "Field organizations shall develop and implement operating procedures for low level waste disposal facilities that ... ensure the security of the facility; minimize the need for long-term control; and meet the requirements of the closure/post-closure plan."	Because of the lack of a design for the disposal facility at this early stage of the project, only limited analyses of the effect of facility degradation were performed. The performance assessment submitted under DOE Order 5820.2A will use the design for the facility and analyzed the consequences of facility degradation.
Other orders	3a(1) Performance Objectives "Protect public health and safety in accordance with standards specified in applicable EH Orders and other DOE Orders"	No specific performance objective resulted from this requirement
Protect air resources	3a(2) Performance Objectives "... Release to the atmosphere shall meet the requirements of 40 CFR 61. ..."	40 CFR 61H and 40 CFR 61Q were used resulting in performance objectives for radon release (<20 pCi m^{-2} s^{-1}) and other radionuclides (10 mrem in a year).

Table D-1. Comparison Between 10 CFR 61 and DOE Order 5820.2A, Chapter III.

Protection of ground water (61.50(a)(4,5,6,7,8))	3a(4) Performance Objectives "Protect ground water resources, consistent with Federal, State, and local requirements"	40 CFR 141 (Federal Drinking Water Standards) was applied to water drawn from a well, resulting in performance objectives for radon concentration (< 3 pCi/ℓ), alpha emitter concentration (< 15 pCi/ℓ), and dose from beta/gamma emitters (< 4 mrem in a year). The time of compliance is 10,000 years and the location of the well is 100 meters downgradient of the disposal facility.
Protection of surface water (61.50(a)(5,6,8))		The State of Washington protection for the Columbia River is more restrictive than 40 CFR 141 for beta/photon emitters. Therefore the dose limit was reduced to 1 mrem/yr. The point of compliance is a well to groundwater just before the groundwater reaches the Columbia River.

RECORD OF REVISION

WHC-SD-WM-TI-699

Page

(2) Title

Technical Basis for Classification of Low-Activity Waste Fraction from Hanford Site Tanks

CHANGE CONTROL RECORD

(3) Revision	(4) Description of Change - Replace, Add, and Delete Pages	Authorized for Release (5) Cog. Engr.	(6) Cog. Mgr.
0	(7) Initial release-EDT #612662	C. A. Petersen 5/17/96	D. J. Washenfel 5/17/96
1	Document update. Replace all pages. ECN-623790	C. A. Petersen 7/15/96	D. J. Washenfel 7/15/96
2 RS	Document update and add Appendix D. ~~EDT-617830~~ ECN 634662	C. A. Petersen 9/18/96 CAPetersen	D. J. Washenfel 9/18/96 DJWashenfelder

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ENGINEERING DATA TRANSMITTAL

Page 3 of 3

1. EDT 617630

2. To: (Receiving Organization) Storage and Disposal Project	3. From: (Originating Organization) Process Technology	4. Related EDT No.: NA
5. Proj./Prog./Dept./Div.: TWRS	6. Cog. Engr.: C. A. Petersen 73510	7. Purchase Order No.: NA
8. Originator Remarks: For approval and release.		9. Equip./Component No.: NA
		10. System/Bldg./Facility: NA
11. Receiver Remarks:		12. Major Assm. Dwg. No.: NA
		13. Permit/Permit Application No. NA
		14. Required Response Date:

15. DATA TRANSMITTED

(A) Item No.	(B) Document/Drawing No.	(C) Sheet No.	(D) Rev. No.	(E) Title or Description of Data Transmitted	(F) Approval Designator	(G) Reason for Transmittal	(H) Originator Disposition	(I) Receiver Disposition
1	WHC-SD-WM-TI-699	-	2	Technical Basis for Classification of Low-Activity Waste Fraction from Hanford Site Tanks	NA	1,2	1	

16. KEY

Approval Designator (F)	Reason for Transmittal (G)	Disposition (H) & (I)
E, S, Q, D or N/A (see WHC-CM-3-5, Sec.12.7)	1. Approval 2. Release 3. Information 4. Review 5. Post-Review 6. Dist. (Receipt Acknow. Required)	1. Approved 2. Approved w/comment 3. Disapproved w/comment 4. Reviewed no/comment 5. Reviewed w/comment 6. Receipt acknowledged

17. SIGNATURE/DISTRIBUTION (See Approval Designator for required signatures)

(G) Reason	(H) Disp.	(J) Name	(K) Signature	(L) Date	(M) MSIN	(J) Name	(K) Signature	(L) Date	(M) MSIN	(G) Reason	(H) Disp.
1	1	Cog.Eng	[signature]	9/18/96	H5-27						
1	1	Cog. Mgr	[signature]	9/18/96	H5-27						
		QA									
		Safety									
		Env.									

18. C. A. Petersen [signature] 9/18/96 Signature of EDT Originator Date	19. G. F. Williamson [signature] 9-18-96 Authorized Representative for Receiving Organization Date	20. Dale Washington [signature] 9/18/96 Cognizant Manager Date	21. DOE APPROVAL (if required) Ctrl. No. ☐ Approved ☐ Approved w/comments ☐ Disapproved w/comments

Verification Required	ENGINEERING		CONSTRUCTION			
[] Yes	Additional	[] $	Additional	[] $	Improvement	[]
[X] No	Savings	[] $	Savings	[] $	Delay	[]

19. Change Impact Review: Indicate the related documents (other than the engineering documents identified on Side 1) that will be affected by the change described in Block 13. Enter the affected document number in Block 20.

SDD/DD	[]	Seismic/Stress Analysis	[]	Tank Calibration Manual	[]
Functional Design Criteria	[]	Stress/Design Report	[]	Health Physics Procedure	[]
Operating Specification	[]	Interface Control Drawing	[]	Spares Multiple Unit Listing	[]
Criticality Specification	[]	Calibration Procedure	[]	Test Procedures/Specification	[]
Conceptual Design Report	[]	Installation Procedure	[]	Component Index	[]
Equipment Spec.	[]	Maintenance Procedure	[]	ASME Coded Item	[]
Const. Spec.	[]	Engineering Procedure	[]	Human Factor Consideration	[]
Procurement Spec.	[]	Operating Instruction	[]	Computer Software	[]
Vendor Information	[]	Operating Procedure	[]	Electric Circuit Schedule	[]
OM Manual	[]	Operational Safety Requirement	[]	ICRS Procedure	[]
FSAR/SAR	[]	IEFD Drawing	[]	Process Control Manual/Plan	[]
Safety Equipment List	[]	Cell Arrangement Drawing	[]	Process Flow Chart	[]
Radiation Work Permit	[]	Essential Material Specification	[]	Purchase Requisition	[]
Environmental Impact Statement	[]	Fac. Proc. Samp. Schedule	[]	Tickler File	[]
Environmental Report	[]	Inspection Plan	[]	other	[X]
Environmental Permit	[]	Inventory Adjustment Request	[]		[]

20. Other Affected Documents: (NOTE: Documents listed below will not be revised by this ECN.) Signatures below indicate that the signing organization has been notified of other affected documents listed.

Document Number/Revision	Document Number/Revision	Document Number/Revision	Revision
NA			

21. Approvals

	Signature	Date		Signature	Date
Design Authority			Design Agent		
Cog. Eng. C. A. Petersen	C A Petersen	9/18/96	PE		
Cog. Mgr. D. J. Washenfelder	D J Washenfelder	9/18/96	QA		
QA			Safety		
Safety			Design		
Environ.			Environ.		
Other			Other		

DEPARTMENT OF ENERGY

Signature or a Control Number that tracks the Approval Signature

ADDITIONAL

Table D-2. Performance Requirements Comparisons.

NRC: 10 CFR 61	DOE: Order 5820.2A Chapter 3
Section 61.56(a)(1) Waste must not be packaged for disposal in cardboard or fiberboard boxes.	Section 3.i.5.a Waste must not be packaged for disposal in cardboard or fiberboard boxes, unless such boxes contain stabilized waste with a minimum of void space.
Section 61.56(a)(2) Liquid waste must be solidified or packaged in sufficient absorbent material to absorb twice the volume of the liquid. Section 61.56(a)(3) Solid waste containing liquid shall contain as little free standing and non-corrosive liquid as is reasonably achievable, but in no case shall the liquid exceed 1% of the volume. Section 61.56(b)(2) ...liquid wastes, or wastes containing liquid, must be converted into a form that contains as little free standing and non-corrosive liquid as is reasonably achievable, but in no case shall the liquid exceed 1% of the volume of the waste when the waste is in a disposal container designed to ensure stability, or 0.5% of the volume of the waste for waste processed to a stable form.	Section 3.i.5.b Liquid wastes, or wastes containing free liquid, must be converted into a form that contains as little free standing and noncorrosive liquid as is reasonably achievable, but, in no case, shall the liquid exceed 1% of the volume of the waste when the waste is in a disposal container, or 0.5% of the volume of the waste processed to a stable form.
Section 61.56(a)(4) Waste must not be readily capable of detonation or reaction at normal pressures and temperatures, or of explosive reaction with water.	Section 3.i.5.c Waste must not be readily capable of detonation or explosive decomposition or reaction at normal pressures and temperatures, or of explosive reaction with water.
Section 61.56(a)(5) Waste must not contain, or be capable of generating, quantities of toxic gases, vapors, or fumes harmful to persons, transporting, handling, or disposing of the waste.	Section 3.i.5.d (Wording identical)

ENCLOSURE 2