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Best-Basis Inventory Template Compositions of Common Tank Waste Layers

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BEST-BASIS INVENTORY TEMPLATE COMPOSITIONS OF COMMON TANK WASTE LAYERS

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EXECUTIVE SUMMARY

The Best-Basis Inventory (BBI) Program is chartered to develop a Best-Basis Inventory (Standard Inventory) of chemicals and radionuclides stored in the 177 underground tanks. These inventories are developed directly from tank sample data or in lieu of sample data, from engineering estimates and the Hanford Defined Waste (HDW) model (Agnew et al. 1997a). One of the primary methods for engineering estimates is to determine the volume of waste layers in a tank and combine these volume estimates with the known compositions of the tank waste layers. These tank waste layer compositions are commonly referred to as waste type templates. The original sample-based waste type template compositions were developed in 1996. These templates only addressed solid wastes (sludges and saltcakes). During subsequent annual updates of the BBI, these compositions were found to be deficient for a variety of reasons. This report documents the derivation of updated waste type templates from tank sample data and HDW model data. Both solid waste type templates and liquid waste type templates have been developed.

Solid waste type templates are based on sample data supplemented with HDW model data. The sample-based portion of the waste type templates are developed from a review of tanks for which samples have been taken and waste layers identified in core profiles and in the HDW model (Agnew et al. 1997a). The determination of whether a tank, or discrete core segments from that tank, are included in a waste type template is based on an assessment of the process knowledge, whether or not a sampling device intercepted the layer of interest, and the potential for adulteration or commingling of the sample by other layers or waste types. In general, the

minimum possible waste layer thickness that can be sampled is a half-length of a core segment (24.1-cm). In practice, because of contamination from adjacent waste layers, a waste layer several half segments thick is often needed to get a representative sample of a tank waste layer. HDW model data is used to establish the concentration of analytes (primarily radionuclides) for which sample data are not available.

Twenty-two different solid waste type templates were developed during this study. Of the 34 million gallons of single-shell tank waste, these waste type templates are applicable to 87 volume percent of the saltcake and 94 volume percent of the sludge. The double-shell tanks have been extensively sampled. Because of this, there is very little benefit to be gained in development of waste type templates for use in estimation of double-shell tank (DST) waste inventories. The BBI inventory consists of 25 chemical species and 46 radionuclide species. In general, for the waste type templates, sample data is the direct basis for 23 of the chemicals, up to eight of the radionuclides, and density and water content. Sample data and process knowledge were used to calculate 16 additional isotopes of U, Pu, Am, and Cm. The remainder of the chemicals and radionuclides are established by process knowledge and modeling.

The solid waste type templates are based on the analysis of solids fraction of core sample segments and are considered to be indicative of waste layers drained of free flowing liquid.

There is little liquid sample data available from single-shell tanks on a segment-by-segment basis. Because of this lack of data, the liquid waste type does not incorporate sample data.

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LIST OF GENERAL TERMS

ANOVA	a nested analysis of variance model
BBI	Best-Basis Inventory
BBIM	Best-Basis Inventory Maintenance tool
cm	centimeters
CW	cladding waste
CWP	PUREX cladding waste
CWR	REDOX cladding waste
DST	double-shell tank
ft	feet
g/L	grams per liter
g/mL	grams per milliliter
HDW	Hanford Defined Waste model
HLW	high-level waste
in.	inches
ITS	In-Tank Solidification
kgal	kilogallons
kL	kiloliters
kW	kilowatts
M	molarity or moles per liter
MTU	Metric Ton Uranium
MW	metal waste
PCA	Principal Component Analysis (a multivarient statistical technique, aka cluster analysis)
PUREX	plutonium-uranium extraction
REDOX	reduction-oxidation
SST	single-shell tank
TCD	Tank Characterization Data
TCR	Tank Characterization Report
TIC	Total Inorganic Carbon
TLM	Tank Layer Model
TOC	Total Inorganic Carbon
TRU	transuranic
UR	uranium recovery
vol%	volume percent
wt%	weight percent
μCi/g	microcuries per gram
μg/g	micrograms per gram

LIST OF WASTE TYPE TERMS

Waste type definitions used in the following discussion are summarized below. These terms are consistent with the terms used in the HDW model.

Waste Type	Definition
1C	$BiPO_4$ First cycle decontamination waste (1944-1956)
2C	$BiPO_4$ Second cycle decontamination waste (1944-1956)
224	Lanthanum fluoride process "224" waste (1952-1956)
1CFeCN	Ferrocyanide sludge from in-farm scavenging of 1C supernatants in TY-Farm (1955-1958)
A1SltCk	Saltcake from the first 242-A Evaporator campaign (1977-1980), same as SMMA1
AR	Washed PUREX sludge (1967-1976)
B	High-level acid waste from PUREX processed at B-Plant for Sr recovery (1967-1972)
BL	Low-level waste from B-Plant Sr and Cs recovery operations (1967-1976)
BSltCk	Saltcake from 242-B Evaporator operation (1951-1953)
BYSltCk	Saltcake from in-tank solidification (ITS) in BY-Farm (1965-1974)
CEM	Portland Cement, added only to tank 241-BY-105
CSR	Cesium recovery, supernatant from which Cs has been removed.
CWP1	PUREX cladding waste (1956-1960)
CWP2	PUREX cladding waste (1961-1972)
CWR1	REDOX cladding waste, aluminum clad fuel (1952-1960)
CWR2	REDOX cladding waste, aluminum clad fuel with some Zr clad fuel (1961-1972)
CWZr1	PUREX zirconium cladding waste, ((1968-1972)
DW	Decontamination waste, primarily from T-Plant
HS	Hot semi-works ^{90}Sr recovery waste (1962-1967)
MW	$BiPO_4$ process metal waste (1944-1956)
OWW3	PUREX organic wash waste (1968-1972)
P1	PUREX high-level waste (1956-1962)
P2	PUREX high-level waste (1963-1967)
P2'	PUREX acid waste to B-Plant (1964-1972)
P3	PUREX high-level waste (1983-1988)
PFeCN1	Ferrocyanide sludge from in-plant scavenged supernatant (1954-1955)
PFeCN2	Ferrocyanide sludge from in-plant scavenged supernatant (1955-1958)
R1	REDOX high-level waste (1952-1958)
R2	REDOX high-level waste (1959-1966)
RSltCk	Saltcake from self-concentration in S- and SX-Farms (1952-1966)
S1SltCk	Saltcake from the first 242-S Evaporator campaign using 241-S-102 feed tank (1973-1976), same as SMMS1

Waste Type	Definition
S2SltCk	Saltcake from the second 242-S Evaporator campaign using 241-SY-102 feed tank (1977-1980), same as SMMS2
SRR	Strontium recovery waste from sluiced PUREX sludge
T1SltCk	Saltcake from the 242-T Evaporator (1951-1955)
T2SltCk	Saltcake from the last 242-T Evaporator campaign (1965-1976)
TBP	Tributyl phosphate process waste (1952-1958), same as UR
TFeCN	Ferrocyanide sludge from supernatants scavenged in CR Vault (1955-1958)
TH1	Thoria (1966)
TH2	Thoria (1970)
UNK	Unknown waste type, source not identified by the Tank Layer Model

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1.0 INTRODUCTION

The chemical processing of irradiated reactor fuel generated large amounts of waste that required storage and disposal. Most of this waste now exists as either a metal oxy-hydroxide sludge or as a saltcake consisting primarily of nitrate, nitrite, phosphate, carbonate, and sulfate salts of sodium. The waste is stored in 177 underground tanks. For various reasons, waste streams from the processing facilities were segregated within the tanks. This practice resulted in the layering of sludges and saltcake. Each process generated unique layer compositions. Knowledge of the composition of these layers and their location from tank farm records can be used to estimate tank waste inventories for tanks where sampling data is not available.

The purpose of this study is to develop waste type templates for use in estimating tank waste inventories for tanks that have not been sampled, or when sample data for a tank is incomplete. Waste type templates are also used as a reference in confirming or rejecting the process history of a waste tank by comparing the tank sample data to waste type template compositions.

The Best-Basis Inventory (BBI) uses both solid waste type templates and liquid waste type templates to establish the contents of tanks which have not been sampled or tanks for which sampling is incomplete. The waste type templates were created from sample data, process knowledge, and Hanford Defined Waste (HDW) model estimates. The original waste type template compositions were developed in 1996 and were limited to analytes for which sample data existed (primarily the waste chemicals). These templates only addressed solid wastes (sludges and saltcakes). During subsequent annual updates of the BBI, these compositions were found to be deficient for a variety of reasons. This report documents the derivation of updated waste type templates from tank sample data and HDW model data. Both solid waste type templates and liquid waste type templates have been developed.

Solid waste type templates are based on sample data supplemented with HDW model data. The solid waste type templates are considered to be indicative of waste layers drained of free flowing liquid. Liquid waste type templates are primarily based on HDW model predictions because adequate liquid sample data were not available.

This report documents how the waste type templates were developed from sample and HDW model data. Discussion of the sample data used in constructing the solid waste type templates constitutes the majority of this report.

1.1 SOLID WASTE TYPE TEMPLATES

The sample based compositions of solid waste type templates were developed from a review of tanks for which samples have been taken and waste layers identified in the HDW model (Agnew et al. 1997a). The decision as to whether a tank, or discrete core segments from that tank, is included in a solid waste type template composition is based on an assessment of the process knowledge, whether or not a sampling device intercepted the layer of interest, and evidence of contamination or adulteration from adjacent tank waste layers.

The primary process knowledge resources used in this assessment are the HDW waste transaction files (Agnew et al. 1997a and 1997b), the tank waste histories (Anderson 1990), process flowsheets for key analytes, and the tank specific BBI evaluations contained in the tank characterization reports (TCRs). The sample data used in this report was extracted from the Tank Characterization Database (TCD) and TCRs.

The waste type templates are generally based on the analysis of solids fraction of core sample segments and, therefore, represent the solids fraction of the waste type layer. The waste type templates are considered to be indicative of waste layers drained of free flowing liquid.

In general, the contribution of sample data to the solid waste type templates is limited by the availability of sample data to 23 chemical species, and at most 8 radionuclides. The chemical species include Al, Bi, Ca, Cl, total inorganic carbon (TIC) as CO_3, Cr, F, Fe, K, La, Mn, Na, Ni, NO_2, NO_3, Pb, PO_4, Si, SO_4, Sr, total organic carbon (TOC), U, and Zr. The radionuclides include total alpha, ^{60}Co, ^{90}Sr, ^{137}Cs, ^{154}Eu, ^{155}Eu, $^{239/240}Pu$, and ^{241}Am. Sixteen radionuclides (isotopes of U, Pu, Am, and Cm) were calculated from U, $^{239/240}Pu$, ^{241}Am, and total alpha sample data and process knowledge. Estimation of the concentrations of Hg and other radionuclides for the solid waste type templates is based on HDW model data and process knowledge.

Sections 2.0, 3.0, and Section 4.1 address the development of the solid waste type templates. Appendix A documents the review of tank sample data used to identify sample compositions that contribute to the templates. Appendix B lists the spreadsheets used to calculate the sample-based portion of the solid waste type templates and identifies the fileserver where they are stored.

1.2 LIQUID WASTE TYPE TEMPLATES

The liquid waste type template compositions are based primarily on HDW model data. Section 4.2 addresses the development of liquid waste type templates.

2.0 SELECTION AND REDUCTION OF SAMPLE DATA

The following methodology was used to develop sample-based compositions from tank sample data for inclusion in the solid waste type template.

The probable locations of tank waste layers were identified from the Tank Layer Model (TLM) (Agnew et al. 1997a). The corresponding samples (typically core segments) were examined for the presence or absence of characteristic analytes identified from process flowsheets. Samples with a matching process history and chemical composition were selected for potential incorporation into a solid waste type template.

Sample results at or below the detection limit were used without the less-than sign, set to zero, or were excluded based on their consistency with detected values from other samples and/or process flowsheets. The approach for evaluation of detection limit values is an established BBI protocol. In certain cases, individual segment values were discarded because of contamination by specific components from adjacent waste layers.

The primary method of obtaining tank solid samples is by core sampling. Each core sample consists of a number of core segments. Analytical results are generally reported core segment or half core segment. Because some tanks contained thicker layers than other tanks, simple averaging of segment values could bias the composition of the waste type template toward a single tank. To minimize this bias, the following procedure was used. Segment values were first weight averaged by tank to remove biases of full versus half segments and cases when multiple samples came from the same riser. Then, the average from each tank was combined with the averages from other tanks by simple averaging. Radionuclide values were decayed to January 1, 1994 prior to the averaging of tank sample values.

The tank waste layer compositions from each of the tanks was compared to tank waste layer compositions from all of the other tanks using Principal Component Analysis (PCA), a multivarient statistical technique. This data set encompassed all of the solid waste templates. Tank waste layer compositions that did not group with other tank waste layer compositions with the same waste type designation were considered to be outliers. All outlier tank waste layer compositions were reviewed. As a result of this review, some of the tank waste layer compositions were rejected from the waste type template composition due to overall inconsistency with the rest of the template data set.

In certain cases, the value for an analyte within a template was found to be based on a single sample or a sample from a single tank. These values were compared to values for the analyte predicted by the HDW model or by process flowsheets. If the sample-based value was found to be substantially inconsistent with the value extracted from the HDW model or the process flowsheet, the sample-based value was excluded from the template. The intent of this review was to ensure that a template would not be biased by a single assay value that was contaminated by another waste type present in the tank.

3.0 SAMPLE BASED SOLID WASTE TYPE COMPOSITONS

Twenty-two different solid waste type templates were developed during this study. These waste type templates are typical of the waste layers found in the single-shell tanks (SSTs). Of the 34 million gallons of single-shell tank waste, these waste type templates are applicable to 87 vol% of the saltcake and 94 vol% of the sludge. No effort was made to develop sample-based waste type templates for double-shell tanks (DSTs). The DSTs have been extensively sampled, and there is little need for waste type templates in estimating the DST inventories. The following sections present each of the solid waste type templates developed for the five chemical processes that generated the bulk of the Hanford Site tank waste.

3.1 BISMUTH PHOSPHATE PROCESS WASTES

The Bismuth Phosphate ($BiPO_4$) process was the first process to separate plutonium from irradiated nuclear reactor fuel. The $BiPO_4$ process operated from 1944 to 1956 in T-Plant and from 1945 to 1952 in B-Plant. B-Plant and T-Plant processed 7,800 MTU during these campaigns. The resulting waste streams were neutralized and sent to underground tanks for storage, or in the case of certain supernatants, routed through the tanks before being transferred to cribs for ground disposal (Kupfer et al. 1999, Section 4.1.1). The original T-Plant tank farms receiving $BiPO_4$ waste were the 241-T, 241-U, 241-TX, and 241-TY Farms. Tank farms receiving $BiPO_4$ waste from B-Plant were the 241-B, 241-C, 241-BX, and 241-BY Farms.

The mission of the $BiPO_4$ process was to recover plutonium from irradiated aluminum-clad natural uranium fuel elements. The $BiPO_4$ process was a batch precipitation process that obtained separated and purified plutonium by a series of carrier precipitations followed by centrifugation and re-dissolving the precipitated cake. Aluminum cladding was dissolved from the irradiated fuel with caustic (sodium hydroxide) creating coating removal waste that was discharged to underground storage tanks. The uranium metal fuel was then dissolved in nitric acid for extraction. Sulfuric acid was added to the metal solution to prevent the subsequent precipitation of uranium, and sodium nitrate was added to insure that plutonium was in the +4 oxidation state. Bismuth nitrate ($BiONO_3$) and phosphoric acid were added to the solution to carrier-precipitate a plutonium (IV) bearing $BiPO_4$ cake. Centrifuges separated the liquid, which contained most of the uranium and fission products from the precipitated cake. The centrifuge cake was purified by two decontamination cycles. Each cycle consists of two $BiPO_4$ precipitations; the first from a Pu(VI) solution so that the Pu remains in solution, and the second from a Pu(IV) solution so that the Pu is carried on the $BiPO_4$. In each decontamination cycle, the centrifuge cake is dissolved with nitric acid, oxidized with sodium dichromate, re-precipitation with $BiPO_4$, and then separated in centrifuges. Final purification was made with a series of lanthinum fluoride (LaF_3) precipitations. Waste streams generated by the $BiPO_4$ process were neutralized with sodium hydroxide and sent to underground storage tanks.

The major $BiPO_4$ process waste streams included coating (cladding) waste (CW) containing the dissolved aluminum cladding, metal waste (MW) containing the uranium and about 90 percent of the fission products, first cycle decontamination (1C) waste containing about 90 percent of the remaining fission products, second cycle decontamination (2C) waste containing essentially all of the remaining fission products, and final plutonium purification/concentration facility (224) wastes that contained low concentration transuranic (TRU) wastes.

The MW, 1C, and 2C $BiPO_4$ process wastes were routed to underground tank "cascades." These are a series (usually three) of connected tanks that overflowed from one to another. The first tanks in the cascade collected the bulk of the solids. The cascade settling process, therefore, had the effect of clarifying the supernatant from the chemicals and radionuclides in the inlet slurry. Because of the cascading, the MW, 1C, and 2C waste solids including the great bulk of the fission products settled and remained in the tanks. The overflow supernatants from the 1C and 2C wastes were routed to cribs or, beginning in 1951, they were transferred to the 242-B or 242-T Evaporators for concentration. The condensates were disposed to ground and the concentrates were sent back to the tank farms for storage as salt slurry or saltcake. The MW supernatants were not cribbed. The 224 wastes were initially considered "disposed" because they were initially routed to concrete settling tanks (241-361) that overflowed to dry wells and later cribs. Later, the T-200 and B-200 tanks were used to settle solids, initially as single tanks and later as cascades.

Five solid waste type templates were developed for the $BiPO_4$ process. These are metal waste (MW), first cycle (1C) waste, second cycle (2C) waste, pre-1949 224 waste, and post-1948 224 waste. The subdivision of 224 waste into pre-1949 and post-1948 is based on differences observed in the waste assays and process records. These differences are discussed further in Section A.1.4 of Appendix A. The sample-based compositions of these solid waste type templates are shown in Table 3-1.

Table 3-1. Bismuth Phosphate Process Solid Waste Type Template Compositions. (2 Sheets)

Analyte	Units	Metal Waste (MW)	First Cycle Waste (1C)	Second Cycle Waste (2C)	Pre-1949 224 Waste	Post-1948 224 Waste
Al	μg/g	0	15,500	2,010	2,050	141
Bi	μg/g	306	18,700	20,500	110,000	43,200
Ca	μg/g	952	768	1,270	6,720	1,080
Cl	μg/g	1,180	1,050	1,050	1,370	740
TIC as CO_3	μg/g	16,700	5,730	10,300	9,880	7,670
Cr	μg/g	2,360	830	1,280	4,370	3,430
F	μg/g	378	9,530	1,920	5,450	6,660
Fe	μg/g	5,340	10,900	16,200	12,300	5,210
K	μg/g	784	342	679	6,530	6,420
La	μg/g	86.1	3.92	45.4	20,000	11,200
Mn	μg/g	881	84.2	88.7	33,600	14,000
Na	μg/g	73,500	94,400	83,000	35,300	32,500

Table 3-1. Bismuth Phosphate Process Solid Waste Type Template Compositions. (2 Sheets)

Analyte	Units	Metal Waste (MW)	First Cycle Waste (1C)	Second Cycle Waste (2C)	Pre-1949 224 Waste	Post-1948 224 Waste
Ni	μg/g	129	20.8	56.4	555	191
NO_2	μg/g	18,900	9,310	10,400	598	562
NO_3	μg/g	43,000	116,000	107,000	48,800	60,600
Pb	μg/g	573	175	913	786	256
PO_4	μg/g	4,490	71,800	33,200	15,900	7,270
Si	μg/g	1,960	7,030	7,980	11,000	1,440
SO_4	μg/g	1,750	12,900	11,400	312	744
Sr	μg/g	48.6	145	261	1,020	481
TOC	μg/g	940	617	1,530	1,330	822
U_{TOTAL}	μg/g	75,000	1,990	1,460	156	183
Zr	μg/g	51.2	129	7.61	30.3	2.89
^{14}C	μCi/g	NR	0.00023	NR	NR	NR
^{90}Sr	μCi/g	1.33	6.09	34.6	1.07	1.85
^{99}Tc	NR	NR	0.0243	NR	NR	NR
^{137}Cs	μCi/g	88.2	23.7	56.2	0.405	0.0394
^{154}Eu	μCi/g	NR	0.00299	NR	NR	NR
^{155}Eu	μCi/g	NR	0.00342	NR	NR	NR
$^{239/240}Pu$	μCi/g	0.0138	0.0903	0.116	NR	NR
^{241}Am	μCi/g	NR	NR	0.587	NR	NR
Total Alpha	μCi/g	1.18	0.129	0.22	1.03	0.243
Density	g/mL	1.78	1.41	1.27	1.27	1.21
% Water	wt%	22.3	57.5	60.2	61.2	74.6

Notes:

NR = not reported

3.2 URANIUM RECOVERY PROCESS AND SCAVENGING PROCESS WASTES

The $BiPO_4$ process did not recover uranium from the dissolved reactor fuel. Metal waste (MW) containing most of the uranium and about 90 percent of the fission products was stored in underground SSTs. The Tributyl Phosphate process was used to recover uranium from the MW sluiced from the tanks. This waste is also referred to as uranium recovery or UR waste in the HDW model. Flowsheet and process history documents, however, usually refer to recovery of uranium at U-Plant as the Tributyl Phosphate Process or the TBP Process. U-Plant has also been referred to as the TBP Plant. The Tributyl Phosphate process began operation in 1954 and ceased operation in 1958.

The mission of the TBP process was to recover uranium from the MW sluiced from the tanks. It is estimated that 89 to 94 percent of the uranium contained in MW was recovered by sluicing (MacCready 1957). Process vaults containing stainless steel vessels were constructed near each tank farm and sluice pits were added to each tank containing MW. The tanks containing MW were each sluiced to the process vaults using their own supernatant. There the solids were re-dissolved with nitric acid and the resultant solution/slurry was pumped to U-Plant, which housed the TBP process. In U-Plant, centrifuges were used to clarify the feed. The MW solution was fed to a solvent extraction system for separation. The solvent extraction system used an organic phase consisting of TBP in a hydrocarbon diluent as the extractant to recover uranium for reuse. Solvent extraction was carried out in a series of three mechanically pulsed solvent extraction columns. The MW solution entered the first column at a feed point intermediate between the top and bottom of the column. The TBP solution entered at the bottom of the column and flowed upwards. At the top of the column, an aqueous scrub stream containing nitric acid was introduced to scrub residual fission products and plutonium from the rising uranium bearing TBP stream. The aqueous scrub stream also contained ferrous ammonium sulfate and sulfamic acid to reduce plutonium and hold it in the relatively inextractable III valence state. The fission products and plutonium remained in the aqueous raffinate from the first solvent extraction column. The organic phase, bearing the uranium, entered a second solvent extraction column in which water was used to strip the uranium from the organic phase. Uranium was recovered from the strip solution by calcination. In the final solvent extraction column, the organic phase was scrubbed with a Na_2SO_4 solution to remove solvent degradation products and remaining fission product impurities. The aqueous raffinate from the process and spent scrub solutions were combined and discharged as TBP waste.

To minimize the need to construct additional tank storage space, scavenging processes were developed to treat TBP wastes stored in the tanks, TBP wastes as they were generated, and 1C waste from T-Plant. These scavenging processes were based on precipitation of $Na_2NiFe(CN)_6$ to selectively precipitate ^{137}Cs from the supernatant. Initially only ^{137}Cs was scavenged. Later, when neutralized supernatants were found to contain high amounts of ^{90}Sr, co-precipitation with $Ca(NO_3)_2$ and nonradioactive $Sr(NO_3)_2$ was used to scavenge ^{90}Sr from solution. Scavenging allowed supernatants to be disposed of to the ground, reducing the volume of waste stored in the tanks. Processing records indicate that Sr usage was limited to in-plant scavenging. Four tanks (241-BY-106, 241-BY-107, 241-BY-108, and 241-BY-110) apparently received all of the Sr used for scavenging.

In-plant ferrocyanide scavenged TBP waste is similar to TBP waste except for increased Ni and Fe content. The HDW model defines two different in-plant scavenged wastes, PFeCN1 and PFeCN2. The PFeCN2 waste stream is distinguished from the earlier PFeCN1 waste by reduced $Na_2NiFe(CN)_6$ content.

In-tank ferrocyanide scavenging of TBP and 1C wastes only treated the tank supernatants. Ferrocyanide scavenging caused the selective precipitation of Cs prior to the disposal of these supernatant to cribs.

Four different solid waste type templates were developed for the Uranium Recovery and Scavenging processes. These are Tri-Butyl Phosphate (TBP) Process waste, in-farm ferrocyanide scavenging of $BiPO_4$ process 1C waste (1CFeCN), in-plant ferrocyanide scavenging of TBP waste (PFeCN), and in-farm ferrocyanide scavenging TBP waste (TFeCN). The sample-based compositions of these solid waste type templates are shown in Table 3-2.

Table 3-2. Tri-Butyl Phosphate Process and Ferrocyanide Scavenging Process Solid Waste Type Template Compositions.

Analyte	Units	Tri-Butyl Phosphate Waste (TBP)	First Cycle Waste Scavenging (1CFeCN)	In-Plant Waste Scavenging (PFeCN)	In-Tank Waste Scavenging (TFeCN)
Al	µg/g	2,210	8,880	31,600	30,100
Bi	µg/g	390	19,400	1,990	NR
Ca	µg/g	3,030	NR	11,300	22,300
Cl	µg/g	1,320	648	2,710	947
TIC as CO_3	µg/g	NR	19,800	28,100	31,500
Cr	µg/g	135	1,410	1,790	275
F	µg/g	0	4,510	2,600	726
Fe	µg/g	21,400	29,900	30,200	22,200
K	µg/g	NR	NR	3,100	596
La	µg/g	0	NR	37.7	52.6
Mn	µg/g	150	1,460	545	214
Na	µg/g	110,000	108,000	158,000	105,000
Ni	µg/g	84.1	1,550	8,800	14,300
NO_2	µg/g	18,100	13,100	42,600	49,100
NO_3	µg/g	186,000	50,300	106,000	58,100
Pb	µg/g	526	596	1,570	2,850
PO_4	µg/g	91,400	72,100	51,400	74,300
Si	µg/g	554	8,790	2,570	4,780
SO_4	µg/g	17,600	3,680	17,700	11,200
Sr	µg/g	592	NR	12,500	356
TOC	µg/g	608	1,770	12,400	2,660
U_{TOTAL}	µg/g	10,900	20,400	30,500	34,700
Zr	µg/g	0	189	24.9	17.0
^{90}Sr	µCi/g	186	110	661	1,350
^{137}Cs	µCi/g	10.4	44.2	414	605
$^{239/240}Pu$	µCi/g	0.0235	0.186	0.111	0.231
^{241}Am	µCi/g	NR	NR	NR	0.248
Total Alpha	µCi/g	0.0292	0.282	0.216	NR
Density	g/mL	1.51	1.65	1.68	1.42
% Water	wt%	44.9	57.7	33.9	39.4

Note: NR = Not reported

3.3 REDOX PROCESS WASTES

In 1952 the world's first radiochemical separations solvent extraction process was put into operation as a replacement for the $BiPO_4$ process. Using Hexone[1] as the extraction solvent, this process offered the advantages of:

- uranium recovery and purification,
- improved plutonium recovery (greater than 98 percent),
- reduced high-level waste volumes,
- increased throughput rates, and
- attendant cost savings.

The process was dubbed the REDOX process (for REDuction-OXidation), the name being descriptive of the plutonium chemistry adjustments which were key in controlling the selective solvent extraction of Pu and effecting its separation from uranium. The REDOX process operated from 1952 to 1966 and was housed in the 202-S Building (S-Plant).

Irradiated fuel elements were chemically declad with caustic (for aluminum cladding removal) or with ammonium fluoride (for zirconium cladding removal). The declad metallic uranium "slugs" were then dissolved in nitric acid as was done in the Bismuth Phosphate process. The dissolver solution was then adjusted in acidity to be less than 0.1 M HNO_3 (a safety requirement for compatibility with the Hexone™ organic solvent), and the plutonium was oxidized with sodium dichromate to the hexavalent state to maximize its extraction into the Hexone™ solvent.

The solvent extraction process employed three and four cycles of extraction/stripping for uranium and plutonium, respectively, in order to achieve the required degree of purification from fission products. Aqueous and solvent streams were counter-currently contacted in up to 57-feet tall, unagitated, columns packed with Raschig rings or Pall rings. A solution of aluminum nitrate was added to extraction feed streams and scrub streams to "salt" the aqueous phase and drive the uranium and plutonium to extract into the solvent phase. (Note, that unlike the plutonium-uranium extraction [PUREX] process, nitric acid could not be used as the "salting" agent because Hexone™ reacts violently with nitric acid even at moderate concentrations.)

The REDOX process also required the addition of smaller amounts of sodium dichromate to maintain Pu valence control in extraction columns and the addition of ferrous sulfamate to effect the reduction of plutonium to the inextractable +3 valence in the 1B partitioning column. These addition chemicals along with the fission products from the irradiated fuel were collected in extraction column raffinate streams and concentrated for disposal as high-level waste. (As the REDOX process flowsheet matured, several of the secondary and tertiary cycle raffinate streams were "back-cycled" to provide aluminum nitrate salting agent for reuse in primary extraction cycles, thereby reducing chemical costs and overall waste volume.) Enough sodium hydroxide

[1] A trademark of the Union Carbide Company, Danbury, Connecticut, for methyl isobutyl ketone.

(four moles per mole of aluminum in the waste) was added to make the waste alkaline and solubilize the aluminum before the waste was transferred to the tank farms. For these reasons, high-level waste (HLW) from the REDOX process is rich in Na, Al, Cr, and Fe.

Initially, REDOX processed aluminum-clad natural enrichment uranium fuel; however, beginning in October of 1958, the mission of the REDOX plant was focused on processing the slightly enriched (0.94 initial wt % U-235) "E-metal" fuel that made up approximately 20 percent of the total fuel cycle at the Hanford Site. Compared to the natural enrichment uranium fuels being processed at the PUREX plant, the E-metal fuel was irradiated to a higher exposure level and, therefore, contributed approximately 50 percent more fission products and associated radiolytic decay heat (on a basis of tons uranium processed) to the REDOX high-level waste. This fact, coupled with process modifications to lower the per-ton-uranium overall waste volume, caused REDOX waste to self-boil in underground tanks. During its life, REDOX processed 11,609 MTU of aluminum-clad natural enrichment uranium fuel, 7,852 MTU of slightly enriched "E-metal" fuel, and 245 MTU of Zircaloy®[2] clad fuel.

Minor Details

The REDOX process was also unique in its early use of a head-end permanganate digestion step to oxidize and volatilize ruthenium tetra oxide to provide an additional stage of separation. (Of all the fission product elements, ruthenium was the most prone to co-extract with U and Pu into the Hexone solvent and thus the need for supplementary ruthenium removal.) This operation led to the presence of some manganese dioxide in REDOX waste solids. In later years this permanganate oxidation step was replaced with ozone oxidation/volatilization operations performed on the U and Pu product streams. Other minor oxidizers such as periodic acid and sodium bismuthate were also added.

During a few turbulent years in the early 1960s, the REDOX process added a moving bed, anion-exchange process step to effect additional purification of plutonium from fission products. This operation was abandoned when fire destroyed the moving-bed exchanger facility.

Beginning in about 1959, many modifications (pH adjustments) to the basic solvent extraction flowsheet were made to make possible the recovery of a third product, Np-237, from the irradiated Hanford fuels.

The REDOX solvent, being a single organic compound, was purified from fission products and organic degradation compounds by steam distillation – a process that avoided the generation of a separate waste stream (as occurs in the PUREX process flowsheet). Compared to the PUREX process solvent, however, the Hexone solvent was more flammable and required the use of inert gas (nitrogen) "blanketing" on process vessels containing the solvent. The Hexone solvent was also more prone to loss via volatilization from the process.

Under the original REDOX flowsheet, each of the solvent extraction cycles generated a waste

[2] Zircaloy and Zircaloy-2 are trademarks of Teledyne Wah Chang, Albany, Oregon.

stream that was concentrated, combined with the cladding waste, and routed to the tank farms for storage as "salt waste" in the 241-S and 241-SX Farms. As the REDOX process matured, several of the intermediate cycle wastes were recycled, thereby reducing the plant chemical consumption and increasing the fission product concentration in the wastes. The fission product concentration (and the associated amount of decay heat) eventually became high enough that the wastes self-boiled. This reduced the waste tank volume required for storage, but necessitated that the waste tank design incorporate features to promote controlled boiling in the tanks. In the mid-1950s, cladding waste was routed to storage in separate tanks from the solvent extraction wastes. Also, in the 1950s a route was installed to utilize the 241-U Farm for storage of some REDOX wastes.

REDOX generated several waste types: cladding waste (CWR1 and CWR2), high-level waste (R1 and R2), and a high salt waste (RSltCk). REDOX cladding waste from aluminum clad fuel is divided by the HDW model into two process campaigns, CWR1 and CWR2. The CWR1 was generated by REDOX from 1952 to 1960, and CWR2 was generated from 1961 to 1967. REDOX cladding waste is characterized by a high aluminum content. The CWR2 has a higher Al/Na ratio than CWR1. REDOX cladding waste was frequently intermixed with REDOX high-level waste in the same storage tanks. The HDW model divides the REDOX process high-level waste into two campaigns, R1 from 1952 to 1958 and R2 from 1959 to 1966. This waste is also characterized by a high aluminum content. The R2 has a higher Al/Na ratio than R1. REDOX high level waste is expected to have more iron and nickel than the cladding waste. Frequently, the only means to distinguish between R1 waste and CWR1 is by the ^{90}Sr concentration. The R1 waste has a higher concentration of ^{90}Sr and other fission products than does CWR1.

Solid waste type templates have only been constructed for the R1 and CWR1 waste layers. The families of tanks that contain the R2 and CWR2 waste types are far fewer than the number of tanks that contain R1 and CWR1 waste layers. The similarity in composition of the R1, R2, CWR1, and CWR2 sludges and the thinness of the R2 and CWR2 layers makes the isolation of representative R2 and CWR2 samples difficult.

Radiolytic decay heat from REDOX high-level waste, caused wastes in many of the REDOX high-level waste receiver tanks in 241-S and 241-SX Farm to boil at atmospheric pressure. This self-concentration of supernatants generated the RSltCk waste. Although there is some uncertainty with respect to which tanks reached boiling conditions, the RSltCk layers are expected to be distinguishable from the R1 and R2 wastes by higher Na concentrations and lower ^{90}Sr concentrations in the RSltCk layers. After fuel reprocessing ended at the REDOX Plant, the REDOX D-12 concentrator was used to concentrate waste liquids by evaporation. Over 44,000 kL (11,600 kgal) of tank space was reclaimed by this operation from 1967 to 1972. The source of the waste liquids is incompletely documented, but the records that exist indicate that at least half of the waste came from Battelle Northwest Laboratory, Douglas United Nuclear, and a couple of the tanks in T-Farm. Data on the disposition of these wastes is even less clear; however, the D-12 concentrator may have contributed to the RSltCk volumes reported by the TLM (Agnew et al. 1997a) from 1967 to 1972.

Three different solid waste type templates were developed for the REDOX process. These are early (1952-1958) REDOX high-level waste (R1), early (1952-1960) REDOX cladding waste from aluminum clad fuel (CWR1), and saltcake from self-concentration of REDOX high-level waste supernatants (RSltCk). The sample-based compositions of these solid waste type templates are shown in Table 3-3.

Table 3-3. REDOX Process Solid Waste Type Template Compositions.

Analyte	Units	REDOX HLW (1952-1958) (R1)	REDOX Cladding Waste (1952-1960) (CWR1)	REDOX Saltcake Waste (RSltCk)
Al	µg/g	143,000	125,000	78,200
Bi	µg/g	74.0	525	5.95
Ca	µg/g	530	431	220
Cl	µg/g	2,460	553	5,250
TIC as CO_3	µg/g	15,800	6,950	3,950
Cr	µg/g	3,350	912	4,140
F	µg/g	1,200	1,560	124
Fe	µg/g	3,300	1,530	794
K	µg/g	732	564	740
La	µg/g	23.4	49.9	0
Mn	µg/g	1,860	113	1,090
Na	µg/g	106,000	43,600	182,000
Ni	µg/g	1,520	369	49.9
NO_2	µg/g	38,500	13,000	33,700
NO_3	µg/g	105,000	24,000	293,000
Pb	µg/g	79.3	2,550	56.1
PO_4	µg/g	3,690	5,260	3,120
Si	µg/g	1,440	1,900	1,100
SO_4	µg/g	2,170	672	1,300
Sr	µg/g	276	56.0	134
TOC	µg/g	1,700	1,280	843
U_{TOTAL}	µg/g	4,470	4,820	2,490
Zr	µg/g	73.4	168	16.7
^{90}Sr	µCi/g	625	11.4	NR
^{137}Cs	µCi/g	92.6	41.5	104
$^{239/240}Pu$	µCi/g	NR	NR	NR
^{241}Am	µCi/g	NR	NR	NR
Total Alpha	µCi/g	0.942	0.111	0.235
Density	g/mL	1.75	1.62	1.72
% Water	wt%	24.8	22.6	28.9

Note: HLW = high-level waste

3.4 PUREX PROCESS WASTES

The PUREX solvent extraction process operated from 1956 to 1972 in the 202-A Building (PUREX Plant). PUREX operations were resumed in 1983, but the facility was again shut down in 1989. Aluminum-clad fuels (65,924 MTU) were processed at the PUREX Plant from 1956 to 1972. Some Zircaloy® clad fuels (1,572 MTU) were processed from 1967 to 1972, and Zircaloy® clad fuels (3,890 MTU) were processed exclusively from 1983 to 1989. Essentially all of the wastes from processing the Zircaloy® clad fuels were discharged to only a few tanks. A tank waste layer composition for the Zircaloy® cladding waste was not developed by this study because tanks containing this waste have been thoroughly sampled.

Aluminum clad fuel elements were declad with caustic and dissolved in nitric acid as was done in the $BiPO_4$ and REDOX processes to make a dissolver solution that contained uranium, plutonium, and the fission products. The primary PUREX solvent extraction processes were performed in a series of three mechanically pulsed solvent extraction columns. In the first solvent extraction column, an organic phase containing 30 vol% TBP in normal paraffin hydrocarbon (NPH) extracts uranyl nitrate and Pu(IV) and Pu(VI) nitrates from the dissolver solution, leaving the bulk of the fission products in the aqueous raffinate. The organic phase from the first solvent extraction column is fed to the second solvent extraction column where organic phase is extracted with an aqueous stream containing ferrous sulfamate. Plutonium is reduced to the III valence state and leaves the column in the aqueous phase. The uranium leaves the column in the organic phase and goes to the third solvent extraction column where the uranium is removed with a dilute nitric acid stream. The PUREX process includes additional solvent extraction columns to purify the plutonium and uranium products. The PUREX process was much more efficient than the REDOX process. The nitric acid used as the salting agent at the PUREX Plant could be partially recovered for reuse, unlike the aluminum nitrate used in the REDOX Plant.

Wastes from the PUREX process included both high-level solvent extraction waste and lower-level wastes such as cladding waste, organic wash waste, and cell drainage waste. Prior to 1972 the high-level waste was routed to 241-A and 241-AX Farms. After 1972 high-level waste was routed to the DST farms. The sodium carbonate organic wash wastes were originally routed to so called high-heat "aging waste" tanks, but after sufficient radioactive decay were routed to 241-C Farm for storage, transfer to other tank farms, and subsequent concentration along with the cladding waste and cell drainage waste. In later years, these wastes were routed to DSTs.

The PUREX Plant generated high-level waste streams and cladding waste streams with several different compositions during the operating life of the plant.

The HDW model subdivides the PUREX high-level waste into three different campaigns, P1, P2, and P3. The P1 waste was generated by PUREX from 1956 to 1962, P2 waste was generated from 1963 to 1967, P2' waste from 1968 to 1972, and P3 waste from 1983 to 1988. The HDW model waste transaction files indicate that P1 waste only went to two tanks, 241-A-101 and 241-A-104. The switch from P1 waste to P2 waste coincides with addition of second cycle uranium extraction recycle and sugar denitration of the high-level waste to the PUREX process,

thereby reducing amount of nitric acid and subsequent caustic addition to this waste stream. The P2' waste was sent directly to B-Plant for strontium removal and appears in the HDW model as B-Plant high- and low-level wastes. P3 waste was sent to double-shell aging waste tanks, 241-AZ-101 and 241-AZ-102. Both of these tanks have been sampled. Waste from the P2 era is the only fraction of PUREX high-level waste dispersed over multiple tanks. Although much of the P2 was sluiced for strontium recovery, the HDW model waste transaction files indicate layers of P2 still reside in seven single-shell tanks. Tanks thought to contain P2 waste include 241-A-105, 241-AX-101, 241-AX-103, 241-AX-104, 241-B-110, 241-B-111, and 241-C-104.

PUREX cladding waste from aluminum clad fuel is divided by the HDW model into two compositions, CWP1 and CWP2. CWP1 was generated by PUREX from 1956 to 1960, and CWP2 was generated from 1961 to 1972. These wastes are characterized by a high aluminum content. CWP2 has a higher Al/Na ratio than does CWP1. CWP2 is more likely to be contaminated with Th, Zr, and Strontium recovery waste (SRR) than is CWP1.

Three different solid waste type templates were developed for the PUREX process. These include the PUREX high-level waste from 1963 through 1967 (P2), cladding waste from 1956 through 1960 (CWP1), and cladding waste from 1961 through 1972 (CWP2). The sample-based compositions of these solid waste type templates are shown in Table 3-4.

Table 3-4. PUREX Process Solid Waste Type Template Compositions. (2 Sheets)

Analyte	Units	PUREX HLW (1963-1967) (P2)	PUREX Cladding Waste (1956-1960) (CWP1)	PUREX Cladding Waste (1961-1972) (CWP2)
Al	μg/g	39,000	140,000	133,000
Bi	μg/g	0	1,090	2,440
Ca	μg/g	7,830	6,000	1,230
Cl	μg/g	313	279	489
TIC as CO_3	μg/g	70,100	18,300	31,200
Cr	μg/g	2,080	557	393
F	μg/g	101	1,030	1,460
Fe	μg/g	145,000	20,400	24,500
K	μg/g	NR	1,160	507
La	μg/g	1,470	432	100
Mn	μg/g	5,910	889	1,180
Na	μg/g	130,000	49,800	37,500
Ni	μg/g	9,330	1,790	1,600
NO_2	μg/g	2,240	7,740	12,100
NO_3	μg/g	57,200	8,220	24,900
Pb	μg/g	9,270	1,330	1,680
PO_4	μg/g	31,900	11,700	9,690
Si	μg/g	12,900	17,000	8,810

Table 3-4. PUREX Process Solid Waste Type Template Compositions. (2 Sheets)

Analyte	Units	PUREX HLW (1963-1967) (P2)	PUREX Cladding Waste (1956-1960) (CWP1)	PUREX Cladding Waste (1961-1972) (CWP2)
SO_4	µg/g	29,600	2,020	2,670
Sr	µg/g	1,490	198	53.3
TOC	µg/g	NR	2,170	671
U_{TOTAL}	µg/g	1,650	13,500	616
Zr	µg/g	3,950	432	1,490
^{90}Sr	µCi/g	26,100	606	1,430
^{137}Cs	µCi/g	954	85.6	29.6
$^{239/240}Pu$	µCi/g	3.90	NR	1.52
^{241}Am	µCi/g	19.0	1.02	1.09
Total Alpha	µCi/g	NR	1.22	NR
Density	g/mL	1.79	1.49	1.71
% Water	wt%	16.1	38.6	29.3

Note: HLW = high-level waste

3.5 WASTE CONCENTRATION SALTCAKE WASTES

To minimize the need to construct additional storage tank space, evaporation was routinely used to reduce overall waste volume. Saltcakes are the product of various efforts to reduce waste volume by evaporation. Saltcakes are mostly made up of sodium salts of nitrate, nitrite, carbonate, phosphate, and sulfate. Transition metals, such as iron, manganese, and lanthanum and heavy metals, such as uranium and lead are typically absent. Aluminum, which is amphoteric, can be present at various concentrations. There were four different waste evaporation methods: the atmospheric-pressure evaporators, 242-B and 242-T; the in-tank solidification tanks (241-BY Farm); the self-boiling tanks (241-S, 241- SX, 241-A, and 241-AX Farms); and the vacuum-evaporators (242-A and 242-B). Early evaporation processes were operated at near atmospheric pressure. More recent evaporation processes operated under a vacuum, allowing evaporation to occur at lower temperatures.

B saltcake (BSltCk) is derived from evaporator concentrates produced from the 242-B Evaporator from 1951 to 1954. The process feeds consisted mostly of $BiPO_4$ process and TBP process supernatants. The 242-B (and 242-T) Evaporators were atmospheric-pressure evaporators that operated at relatively high temperatures. These evaporators were essentially steam-heated pots equipped with off-gas condensing and filtering equipment. The high temperatures and high air flows may have contributed to the absorption of CO_2 and the formation of carbonate salts. Tank 241-B-106 was used as the evaporator feed tank, and the evaporator bottoms receiver tanks included 241-B-104, 241-B-105, 241-B-107, 241-B-108, and 241-B-109 (Agnew et al. 1996). These tanks collectively received about 27,400 kL (7,240 kgal) of evaporator concentrates, which produced 1,820 kL (480 kgal) of BSltCk solids (Agnew et al. 1997a).

The in-tank solidification (ITS) process used an in-tank heater in one of the underground storage tanks as an atmospheric-pressure evaporator. The evaporation or boiling tank was connected with a series of receiver tanks that were used to cool the concentrated waste liquids and precipitate salts. This process formed saltcake in the cooling tanks. In 1965, a prototype in-tank heater (ITS-1) was initially installed in tank 241-BY-101 to concentrate the recirculating supernatant from other 241-BY Farm tanks. The ITS-1 Evaporator consisted of a 4,000 kW electric immersion heater installed inside an airlift circulator. Later, in 1966, a second heater (ITS-2) was placed in tank 241-BY-102, with tank 241-BY-103 being used as a primary feed tank. The ITS-2 Evaporator used steam heated tubes inside an airlift circulator to heat the waste. Finally, in 1966, a third heater (ITS-2) was placed in tank 241-BY-112, with tank 241-BY-109 acting as the primary feed tank. During this third evaporation campaign, the heating coils in tank 241-BY-102 were converted to cooling coils and used to cool the hot concentrate from tank 241-BY-112 that was being routed to tank 241-BY-102. Air sparging within the air lift circulators helped circulate the waste and removed water vapor from the tanks. The high contact with air in these processes may have contributed to the absorption of CO_2 and the formation of carbonate salts. Because of equilibrium differences between the boiling tanks and the cooling tanks, the saltcake composition of the cooling tanks probably does not represent the composition of the boiling tanks. The in-tank solidification (ITS) program ended in 1976 with 144,000 kL (38,000 kgal) of volume reduction and about 14,700 kL (3,900 kgal) of saltcake (BYSltCk) being formed. The composition of BYSltCk reported in this document is based on the reported saltcake compositions of the cooling tanks.

The T1SltCk group consists of those tanks that received evaporator bottoms from the first 242-T Evaporator campaign (1951 to 1955). The process feeds consisted mostly of $BiPO_4$ process and TBP process supernatants. The 242-T (and 242-B) Evaporators were atmospheric-pressure evaporators that operated at relatively high temperatures. This may have contributed to the absorption of CO_2 and the formation of carbonate salts. Tank 241-TX-118 was the evaporator feed tank, and the evaporator bottoms receiver tanks included 241-T-109, 241-TX-103, 241-TX-113, 241-TX-116, 241-TX-117, and 241-TY-102. These tanks collectively received some 35,100 kL (9,280 kgal) of evaporator concentrate, with tanks 241-TX-116 and 241-TX-117 receiving more than 80 percent of the total inventory (10,200 kL [2,686 kgal] in tank 241-TX-116 and 18,400 kL [4,871 kgal] in tank 241-TX-117) (Agnew et al. 1996).

T2SltCk waste is comprised of the solids produced from the 242-T Evaporator concentrates during the period from 1965 to 1976. Tank 241-TX-118 was used as the evaporator feed tank for the 242-T Evaporator, and the bottom receiver tanks included one tank from the 241-S Farm, two tanks from 241-SX Farm, four tanks from 241-T Farm, 17 tanks from 241-TX Farm, two tanks from 241-TY Farm, and five tanks from 241-U Farm. These tanks collectively contain about 19,600 kL (5,200 kgal) of T2SltCk solids, with about 77 percent of this inventory (15,100 kL [4,000 kgal]) residing in tanks that have not been sampled in the 241-TX and 241-U Farms.

Four different solid waste type templates were developed for the atmospheric-pressure evaporation processes. These are the 242-B Evaporator saltcake (BSltCk), in-tank solidification saltcake (BYSltCk), 242-T Evaporator saltcake from 1951 through 1955 (T1SltCk), and 242-T Evaporator saltcake from 1965 through 1976 (T2SltCk). The sample-based compositions of these solid waste type templates are shown in Table 3-5.

Table 3-5. Solid Waste Type Template Compositions for Atmospheric-Pressure Evaporation Processes.

Analyte	Units	242-B Evaporator Saltcake (BSltCk)	In-Tank Solidification Saltcake (BYSltCk)	242-T Evaporator Saltcake (1951-1955) (T1SltCk)	242-T Evaporator Saltcake (1965-1976) (T2SltCk)
Al	µg/g	39,200	26,100	1,760	17,600
Bi	µg/g	NR	21.5	388	60.1
Ca	µg/g	1,339	567	251	232
Cl	µg/g	1,257	1,700	623	3,330
TIC as CO_3	µg/g	10,500	66,300	33,900	46,000
Cr	µg/g	1,150	2,240	69.7	1,760
F	µg/g	13,600	10,200	8,930	2,440
Fe	µg/g	1,990	1,030	5,800	1,330
K	µg/g	912	2,160	239	1,120
La	µg/g	NR	NR	39.8	32.0
Mn	µg/g	227	93.0	606	488
Na	µg/g	190,000	216,000	217,000	208,000
Ni	µg/g	42.6	192	15.9	51.8
NO_2	µg/g	7,320	25,700	7,140	26,600
NO_3	µg/g	93,100	236,000	212,000	355,000
Pb	µg/g	309	100	418	157
PO_4	µg/g	108,000	20,300	136,000	15,700
Si	µg/g	2,100	1,300	1,190	920
SO_4	µg/g	77,900	25,100	14,100	25,100
Sr	µg/g	68.2	138	21.6	5.89
TOC	µg/g	739	4,750	NR	3,560
U_{TOTAL}	µg/g	1,900	538	567	915
Zr	µg/g	53.6	15.9	11.6	24.9
^{90}Sr	µCi/g	2.0	17.1	0.115	26.8
^{137}Cs	µCi/g	18.3	85.8	2.62	90.3
$^{239/240}Pu$	µCi/g	NR	NR	NR	NR
^{241}Am	µCi/g	NR	NR	NR	NR
Total Alpha	µCi/g	0.0219	0.0383	0.0420	0.327
Density	g/mL	1.75	1.69	1.55	1.67
% Water	wt%	39.7	28.9	22.5	27.6

The 242-A and 242-S Evaporator/crystalizers operated in a different manner from the atmospheric-pressure evaporators used in previous waste volume reduction efforts. These steam-heated evaporators operated under a partial vacuum (typically 0.05 atmospheres). This reduced the amount of heat required to boil the supernatants and minimized the interactions with air, therefore, reducing carbonate formation.

A1SltCk waste is comprised of the solids produced from 242-A Evaporator during the period from 1976 to 1980. This waste was routed to 26 different receivers in the 241-A, 241-AW, 241-AX, 241-AY, 241-AZ, 241-BX, 241-BY, and 241-C Farms (Agnew et al. 1996). These tanks collectively received about 104,000 kL (27,500 kgal) of A1SltCk concentrate and contain 7,150 kL (1,890 kgal) of A1SltCk solids, with about 38 percent of this inventory (2,736 kL) [723 kgal] residing in tank 241-AX-101, which has not been sampled. Tank 241-A-102 was used as the evaporator feed tank.

S1SltCk concentrates were produced from the 242-S Evaporator from 1972 to 1977. These concentrates were added to a number of bottom receivers, including 10 tanks in the 241-S Farm, six tanks in the 241-SX Farm, eight tanks in the 241-U Farm, and finally tank 241-SY-102. Tank 241-S-102 was used as the evaporator feed tank.

S2SltCk waste is comprised of the solids produced from the 242-S Evaporator during the period from 1977 to 1980. Tank 241-SY-102 was used as the evaporator feed tank. The S2SltCk concentrates were routed to 21 different receivers tanks in the 241-A, 241-S, 241-SX, 241-SY, and 241-U Farms (Agnew et al. 1996).

Three different solid waste type templates were developed for the vacuum evaporation/ crystallization processes. These include the 242-A Evaporator saltcake from 1977 through 1980 (A1SltCk), 242-S Evaporator saltcake from 1973 through 1976 (S1SltCk), and 242-S Evaporator saltcake from 1977 through 1980 (S2SltCk). The sample-based compositions of these solid waste type templates are shown in Table 3-6.

Table 3-6. *Solid Waste Type Template Compositions for Vacuum Evaporation/Crystallization Processes.*

Analyte	Units	242-A Evaporator Saltcake (1977-1980) (A1SltCk)	242-S Evaporator Saltcake (1973-1976) (S1SltCk)	242-S Evaporator Saltcake (1977-1980) (S2SltCk)
Al	µg/g	24,100	17,500	18,600
Bi	µg/g	172	62.0	66.3
Ca	µg/g	434	227	275
Cl	µg/g	5,800	4,000	4,320
TIC as CO_3	µg/g	82,600	42,900	36,300
Cr	µg/g	4,500	4,990	5,300
F	µg/g	693	2,540	592
Fe	µg/g	321	1,330	1,030
K	µg/g	3,400	978	1,460
La	µg/g	63.9	27.1	30.8
Mn	µg/g	54.3	467	203
Na	µg/g	175,000	204,000	192,000
Ni	µg/g	256	92.6	79.6
NO_2	µg/g	85,300	49,000	48,600
NO_3	µg/g	4,500	269,000	258,000
Pb	µg/g	89.4	119	82.4
PO_4	µg/g	15,300	26,200	14,800
Si	µg/g	428	718	1,040
SO_4	µg/g	14,900	13,700	11,000
Sr	µg/g	25.2	15.9	6.18
TOC	µg/g	8,250	4,120	5,460
U_{TOTAL}	µg/g	794	698	371
Zr	µg/g	82.8	31.1	17.5
^{60}Co	µCi/g	0.0415	NR	0.0552
^{90}Sr	µCi/g	41.0	16.1	4.04
^{137}Cs	µCi/g	175	134	179
^{154}Eu	µCi/g	NR	1.14	0.663
^{155}Eu	µCi/g	NR	1.03	0.727
$^{239/240}Pu$	µCi/g	0.704	NR	NR
^{241}Am	µCi/g	0.0793	NR	NR
Total Alpha	µCi/g	0.0635	0.284	0.247
Density	g/mL	1.58	1.65	1.59
% Water	wt%	39.0	32.4	34.1

4.0 IMPLEMENTATION OF WASTE TYPE TEMPLATES

The creation of sample-based template data for the solid waste types provided improved estimates that could be utilized when a tank waste had not been sampled or when laboratory analyses of a tank waste did not include all of the BBI analytes.

To simplify preparation of the BBIs, the sample-based templates, process knowledge estimates, and HDW model predictions were combined to create a concentration vector for each waste type that could be used in the Best-Basis Inventory Maintenance (BBIM) tool. Separate liquid and solid vectors were created for each waste type. Ten additional solid waste type templates were developed directly from HDW model data in the absence of sample data. Several of these templates provide supplemental information with respect to DST wastes that have been sampled. The remainder of these solid waste type templates represent waste types that account for approximately 10 percent of the SST waste volume.

The BBI is maintained by the BBIM tool. The BBIM tool is a Microsoft® Access 2000[3] database application that includes built-in algorithms to calculate the waste inventory of the underground waste tanks from laboratory and engineering data. The concentration values and the density that describe a waste type (or a waste sample) are stored in the BBIM as a vector (Bobrowski et al. 2000).

4.1 SOLID WASTE TYPE BBIM TEMPLATE VECTORS

The BBI templates for the sludge and saltcake solid wastes types are based on sample data representing the waste type, process knowledge estimates for mercury, and Tank Layer Model (TLM) predictions from the HDW model (Agnew et al. 1997a). The combination of the data was performed in a spreadsheet, with precedence given to the sample data representing the solid waste type. The procedure for combination of the data is as follows:

1. The Tank Layer Model (TLM) data for both solid and supernatant wastes were imported electronically into a spreadsheet from the HDW model spreadsheet system.

2. The HDW model saltcake compositions contain both a precipitated salt component and an interstitial liquid component. These compositions were adjusted to reflect a saltcake from which the interstitial liquid had drained. The drainable interstitial liquid was assumed to account for 25 volume percent of the HDW model saltcake. This assumption is consistent with the analysis of Field (2000), which reflects the composition of a tank that has been saltwell pumped and matches the logic used to build BBIs. No adjustments for interstitial liquid were made for the sludge waste types because the sludge interstitial liquid does not usually drain during sample extrusion.

[3] Access 2000 is a registered trademark of Microsoft, Redmond, Washington.

3. Corrections were made to the half-lives of ^{79}Se and ^{126}Sn. The HDW model estimates for these two radionuclides are derived from ORIGEN2 data that included conversions with outdated specific activities.

4. The HDW model included three analyte concentrations that were negative (wt% water for the DE waste type [diatomaceous earth] and OH/free OH for NIT [Partial Neutralization Feed]). These concentrations were set to zero.

5. Sample-based data representing the waste types were imported electronically from the template spreadsheets.

6. The HDW model concentrations were normalized to the sample-based wt% water and density.

7. The sample-based uranium and alpha analytes (Pu, Am, and Cm) were distributed based on the isotopic ratios indicted by the HDW model.

8. The mercury concentrations were revised to reflect the process knowledge estimates for mercury (Higley 2000).

9. The HDW model predictions for ammonia, cyanide, and oxalate were deleted. This data was judged to be unreliable because of the potential release to the tank vapor space, chemical decomposition, or in-growth from degrading organic compounds.

This procedure created a combined vector for the solid waste types with precedence given to the sample-based estimates and the process knowledge estimates for mercury. Gaps in data were supplemented with the adjusted HDW model predictions. Most of the BBI radionuclides are not measured by sample assay and are based on adjusted HDW model predictions.

4.2 LIQUID WASTE TYPE BBIM TEMPLATE VECTORS

The liquid BBI templates are primarily based on the TLM predictions made in the HDW model Rev. 4 (Agnew et al. 1997a). The concentration data for the supernatant waste were imported electronically from the HDW model spreadsheet system. Two modifications to the concentration data were made based on process knowledge or sample data. The mercury concentrations for all supernatant wastes were set to zero based on Higley (2000) process estimates. The concentrations of nine metal analytes that are relatively water-insoluble (Bi, Ca, Fe, La, Ni, Mn, Si, U, and Zr were reduced from the HDW model values based on analytical data available in the TCD. The concentrations of the uranium isotopes were scaled downward to match the revised chemical uranium concentration estimate.

Half-life corrections were also made to the ^{79}Se and ^{126}Sn concentrations. The HDW model predictions for these two radionuclides are derived from ORIGEN2 data that included conversions with outdated specific activities.

4.3 USE OF THE BBIM TEMPLATES VECTORS

Because the moisture content and density of tank wastes can vary widely between tanks containing the same waste type, the BBIM includes algorithms to normalize the solid and liquid template concentrations to the wt% water and density of the tank to which the template vector is being applied. These adjustments are implemented in the BBIM tool (database) using the multiplier function. Provisions were included in the combined template spreadsheets to calculate multipliers for the engineer evaluating the tank by entering the wt% water and density of tank to which the template is to be applied. Further discussion of multiplier usage can be found in Tran (1999).

Liquid concentrations predicted by the HDW model were noted to be unrealistically high for three saltcake waste types (A1SltCk, S1SltCk, and S2SltCk) based on comparison with analytical data. The liquid inventories would be substantially overstated when sample data for wt% water and density adjustment were not available. Standard multipliers were established from a survey of known interstitial liquids for use when the liquid inventory would be entirely based on one of these waste type templates (Place 2001).

The spreadsheets used to calculate the vectors from the merged sample and HDW model data are stored on AP001 fileserver at STDINV\BBI\Waste Template FY01\Solid Template Version 10C.xls and Liquid Template Version 10C.xls. Waste type templates may be modified in the future because of new sample data, and new process knowledge. If changes to the waste type templates are made, the template spreadsheets will be assigned new consecutive revision numbers, and this document will be updated via an Engineering Change Notice (ECN).

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APPENDIX A

EVALUATION OF TANK WASTE LAYER SAMPLE DATA

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APPENDIX A

EVALUATION OF TANK WASTE LAYER SAMPLE DATA

A.1.0 BISMUTH PHOSPHATE PROCESS WASTES

The $BiPO_4$ process was the first process to separate plutonium from irradiated nuclear reactor fuel. The $BiPO_4$ process operated from 1944 to 1956 in T-Plant and from 1945 to 1952 in B-Plant. The resulting waste streams were neutralized and sent to underground tanks for storage, or in the case of certain supernatants, routed through the tanks before being transferred to cribs for ground disposal (Kupfer et al. 1999, Section 4.1.1). The original T-Plant tank farms receiving $BiPO_4$ waste were the 241-T, 241-U, 241-TX, and 241-TY Farms. The original B-Plant tanks farms were the 241-B, 241-C, 241-BX, and 241-BY Farms. There were five major waste streams generated by the $BiPO_4$ process, including: coating waste [cladding waste (CW)] containing the dissolved aluminum cladding, metal waste (MW) containing the uranium and about 90 percent of the fission products, first cycle decontamination (1C) waste containing about 90 percent of the remaining fission products, second cycle decontamination (2C) waste with essentially all of the remaining fission products, and final plutonium purification/concentration facility (224) wastes that contained low concentration transuranic (TRU) wastes.

The $BiPO_4$ process wastes were routed to underground tank "cascades." These are a series (usually three) of connected tanks that overflowed from one to another. The first tanks in the cascade collected the bulk of the solids. The cascade settling process, therefore, had the effect of clarifying the supernatant from the chemicals and radionuclides in the inlet slurry. Because of the cascading, the MW, 1C, 2C, and 224 waste solids including the great bulk of the fission products settled and remained in the tanks. The overflow supernatants (with the exception of MW) were routed to cribs; beginning in 1951, they were transferred to the 242-B or 242-T Evaporators for concentration. Because of the high fission product activity of MW supernatant, these supernatants were stored with the MW sludge. Evaporator condensates were disposed to ground, and the concentrates were sent back to the tank farms for storage as salt slurry or saltcake.

A.1.1 MW TANK WASTE LAYER ASSESSMENT

The MW group consists of those tanks that initially received primary metal waste from the BiPO4 separation process. The MW stream was the first waste stream from the BiPO4 process, and as such, normally contained all of the uranium, 90 percent of the fission products and about 1 percent of the plutonium in the dissolved fuel. The MW composition is aptly described in the Uranium Recovery Manual (GE 1951). The MW analyte concentration estimates were also generated in the HDW model (MW1 and MW2). According to the HDW model, 40 tanks are considered to have a small inventory of MW, mostly in the form of a residual sludge layer varying from 2 to 40 cm thick in the bottom of the tanks. Tanks that are reported by Agnew et al. (1997a) to contain this MW layer include 241-B-101, 241-B-102, 241-B-103, 241-BX-101,

241-BX-102, 241-BX-103, 241-BX-104, 241-BX-105, 241-BX-106, 241-BY-101, 241-BY-102, 241-BY-105, 241-BY-109, 241-BY-111, 241-BY-112, 241-C-101, 241-C-102, 241-C-104, 241-C-201, 241-C-203, 241-C-204, 241-U-101, 241-U-102, 241-U-103, 241-U-104, 241-U-105, 241-U-106, 241-U-108, 241-U-109, 241-U-110, 241-T-101, 241-T-102, 241-T-103, 241-TX-101, 241-TX-102, 241-TX-104, 241-TX-105, 241-TX-106, 241-TX-107, and 241-TX-108. However, Rhodenhizer (1987) contradicts Agnew with respect to how much MW remains in the tanks. In general, the Rhodenhizer document indicates that Agnew et al. (1997a) overestimate the amount of MW that remains in the tanks. The HDW model (Agnew et al. 1997a) was used to generate a set of estimates showing the projected thickness of the MW layer in each tank. These results were then compared to core sample data to determine which tank waste layers might be suitable candidates for inclusion in the MW template. Of the 40 tanks with predicted MW layers, 19 have not been sampled. Most of the sampled tanks were eliminated because the cores did not extend to the bottom of the tank into the zone where MW would be expected. For those tanks where adequate cores were taken, the core segment data was screened to determine which segments contained high uranium concentrations indicative of the MW layer. Based on this analysis, tanks 241-BX-104 and 241-U-105 were found to be the most suitable candidates for the MW template. The limited number of tanks is not surprising because historical reports indicate that more MW was removed from the tanks for uranium recovery than was assumed in the HDW model.

According to the HDW model, tank 241-BX-104 currently contains 155 kL (41 kgal) of MW and 208 kL (55 kgal) of unknown waste. The MW layer was found in the lower half of segment 2 of core 126, based on high uranium content (Hu and Stephans 1996). From the waste transaction records, tank 241-BX-104 received a substantial volume MW in 1954, followed by a small volume of PUREX coating waste (CWP2) from tank 241-C-102 in 1962 and from tank 241-C-108 in 1964. These transfers are important because they help to explain the source of the high aluminum concentration found in the MW layer in tank 241-BX-104. The best estimate of the date of analysis for decay calculations is February 1, 1996.

Tank 241-U-105 was the middle tank in a three-tank cascade that received MW from 1947 through 1956. The MW was sluiced from this tank twice, first in 1953 and then again in 1956. A final MW transfer was received in 1957 after the last sluicing campaign. A significant volume of CWR1 coating waste also was received from tank 241-S-107 via tank 241-U-108 in 1961. Later transfers of 242-T Evaporator concentrates were received from tank 241-TX-118 in 1975-1976, followed by similar transfers of 242-S Evaporator concentrates from tank 241-SY-102 in 1977-1978. Based on this transaction history, tank 241-U-105 is expected to have a mixture of MW and CWR1 waste in the bottom of the tank, with overlying layers of T2SltCk, S1SltCk, and S2SltCk. This profile is consistent with the high aluminum and uranium concentrations found in the lower half of segment 9A of core 136, which suggest a mixture of MW and CWR1 waste in the bottom of the tank (Brown and Franklin 1996). This layer, however, is not uniformly dispersed across the bottom of the tank because uranium concentrations were found to be much lower in several adjacent segments from other cores (cores 131 and 133). The best estimate of the date of analysis for decay calculations is April 1, 1996.

The MW waste type template composition was developed by averaging the analytical data for tanks 241-BX-104 (core 126:2) and 241-U-105 (core 136:9A) for the following analytes: Bi, Ca, Cl, TIC, Cr, F, Fe, La, Mn, Na, NO_2, Pb, PO_4, Si, SO_4, Sr, U, and Zr. Total alpha, ^{241}Am, ^{137}Cs, ^{60}Co, ^{154}Eu, and ^{155}Eu were derived from tank 241-U-105 waste because this was the only source for such information. High aluminum values from both tanks were rejected because these values were probably caused by blending with CWP and CWR waste. Potassium and nickel data from tank 241-U-105 were used because representative data were not available from tank 241-BX-104. The TOC value from tank 241-U-105 was judged to be too high because complexant concentrate wastes were also added to this tank. The TOC value was, therefore, derived from tank 241-BX-104 waste. Finally, high NO_3 values from tank 241-U-105 were rejected because of possible contamination from T2SltCk, S1SltCk, and S2SltCk layers in this tank. The spreadsheet for the MW waste type template composition (MW.xls) is stored on fileserver AP001\STDINV\RPP-8847\MW.xls.

A.1.2 1C TANK WASTE LAYER ASSESSMENT

The $BiPO_4$ process flowsheet compositions of $BiPO_4$ Process first cycle (1C) wastes prior to neutralization are provided in Appendix C of Kupfer et al. (1999). Additional details of the $BiPO_4$ process flowsheet are provided in Schneider (1951). After neutralization, the 1C wastes were combined with aluminum cladding wastes (CW) in the 1C waste tank cascades. The major components in 1C/CW waste (hereafter referred to as 1C waste) are Al, Bi, Ce, Cr, Fe, Na, Si, U, F, NO_3, PO_4, and SO_4. Based on the predicted chemistry of the neutralized waste, the precipitated solids would be expected to consist primarily of Al, Bi, Fe, and Si with some contributions also from the more soluble components such as Na, NO_3, F, PO_4, and SO_4. The bulk of the soluble components would be expected to remain in the supernatants for disposal to ground, or for evaporation to saltcake (e.g., see BSltCk and BYSltCk waste type assessments). The HDW model transaction records and resulting TLM (Agnew et al. 1997a) provide an estimate of the volume of $BiPO_4$ process 1C waste solids deposited in the representative SSTs (primarily tanks in the 241-T, 241-B, and 241-C Farms).

Agnew et al. (1997a) predict eighteen 200 West Area tanks and eleven 200 East Area tanks to contain 1C waste solids. Core samples (1990 and more recent) have been analyzed for fifteen of these tanks. Examination of core sample analytical data at a segment or sub-segment level along with components predicted from the flowsheet indicates that only five of these tanks (241-B-107, 241-BX-107, 241-BX-112, 241-C-110, and 241-T-104) contain what appears to be nearly exclusively 1C sludge solids. The other sampled tanks were considered unsuitable for use in the 1C waste type template composition due to poor or no recovery, or adulteration from other waste types.

The composition of waste in tank 241-B-107 was not included in previous 1C waste type template. However, the recent (1997) sample data indicated that this waste exhibits the characteristics of 1C waste. The waste in tank 241-T-107 was previously included in the 1C tank waste layer composition; however, for reasons explained later in this section, tank 241-T-107 is now excluded.

A synopsis of the composition basis for each of the five tanks containing the 1C $BiPO_4$ process waste type is described below. Agnew et al. (1997a) divide the waste into 1C1 and 1C2 depending on the processing time frame and plant origin. However, the analytical data examined show no apparent differences relating to either process plant origin or process flowsheet. The first cycle $BiPO_4$ process waste type template is, therefore, simply designated as 1C waste.

Two core samples (cores 217 and 218) were taken from tank 241-B-107 in September 1997. Sample recovery from core 217 was good; however, recovery from core 218 was poor (Conner et al. 1998). Core segment compositions were reasonably consistent for core 217 segments 1-3. What is particularly apparent from the two cores, which were taken from opposite sides of the tank, is that core 217 contained primarily 1C waste and core 218 contained primarily BSltCk based on the high sodium and low bismuth concentrations. Core 218 upper half segment 4 consisted mostly of 1C waste; however, it was contaminated with significant amounts of BSltCk. The reconciled data, therefore, exclude all data from core 218. Except for total alpha analysis, no other radionuclide analyses were performed on the core samples.

Two core samples (cores 40 and 41) were recovered from tank 241-BX-107 in 1992 (Winkelman et al. 1997). Each core consisted of seven segments. Segments 1 and 3 of core 40 were not successful because of mechanical failure. Similar problems also occurred for segments 1 through 5 of core 41, although some waste material was recovered from segments 2, 3, and 5 of that core (especially segments 2 and 3). Two separate composites were made, however, from the limited segment material for both cores 40 and 41. Surprisingly, the composition data for the four composites were generally consistent within the tank and also compared well with samples of 1C waste from other tanks. The analysis date of August 1992 was used to decay the radionuclides to January 1, 1994.

According to the HDW model, tank 241-BX-112 currently contains 121 kL (32 kgal) of BSltCk in addition to 500 kL (132 kgal) of 1C waste. A review of the core sample segment data for cores 118 and 119 for tank 241-BX-112 (Winkelman and Morris 1996), indicates that no separate saltcake layer exists although small amounts of saltcake may be intermixed with the 1C sludge. Based on the general absence of any data that would support the presence of a BSltCk layer and the uniform concentrations of 1C sludge components, core sample composite data from this tank were used to represent the composition of 1C waste. For radionuclide decay, an analysis date of January 1, 1996 was used.

Three core samples (cores 37, 38, and 39) were obtained from tank 241-C-110 in 1992 (Benar et al. 1997c). Each core consisted of 4 segments. Two core sample composites were analyzed for cores 37 and 39 and one core composite for core 38 (five total core composites). Composite results were used to represent the composition of 1C in this tank. The limited amount of segment level data supports this choice because the Bi and Ce concentration profiles are generally uniform for these segments (segments 4A and 4B from core 38 and segments 3A and 3B from core 39) and matched the composite data. Radionuclide values were decayed to January 1, 1994, based on the analysis date of June 1992.

Two core samples (cores 45 and 46) were obtained from tank 241-T-104 in 1992 (Sasaki et al. 1997a). Two composite samples were formed from each core. These composite samples appear to be acceptable for the 1C tank waste layer composition because the sample data for the four composites were generally consistent, and compared well with sample data form other 1C waste tanks and with the predicted chemical composition based on the $BiPO_4$ process flowsheet. The analysis date of October 1992 was used for radionuclide decay.

According to Agnew et al. (1997a), tank 241-T-107 contains essentially all 1C waste. Agnew et al. (1997a), however, report nearly 4,900 kL (1.3 Mgal) of TBP Process waste entering the tank in 1953 with possible precipitation of nearly 151 kL (40 kgal) of TBP sludge solids. Three core samples (cores 50, 51, and 52) were taken from this tank in 1992-1993 (Sasaki et al. 1997b). Both segment and composite analyses of the cores indicated that the waste is primarily 1C but is contaminated with significant amounts of TBP waste. The concentrations of iron and uranium, which are key indicators for TBP waste, are up to three fold and twenty fold higher, respectively, than for any of the other candidate 1C waste tanks. In addition, the concentration of bismuth is approximately 30 percent lower than for the other tanks. The analytical data for this tank were, therefore, not selected to represent the 1C waste type template.

The 1C waste type template composition is based on sample values from tanks 241-B-107, 241-BX-107, 241-BX-112, 241-C-110, and 241-T-104. The spreadsheet for the 1C waste type template composition (1C.xls) is stored on fileserver AP001\STDINV\ RPP-8847\1C.xls.

A.1.3 2C TANK WASTE LAYER ASSESSMENT

Second-cycle (2C) wastes were also generated in the $BiPO_4$ separation process. This waste typically contained about 0.1 percent of the fission products and 1 percent of the plutonium from dissolved fuel. Second-cycle supernatant was usually routed to cribs. According to the HDW model, 2C waste was transferred to tanks 241-B-104, 241-B-105, 241-B-110, 241-B-111, 241-B-112, 241-T-105, 241-T-110, 241-T-111, and 241-T-112. Sample data were not available for tanks 241-B-105, 241-B-112, 241-T-105, and 241-T-110.

Tank 241-B-104 received 8,656 kL (2,289 kgal) of 2C1 waste from 1945-1950; 15,750 kL (4,161 kgal) of 2C2 waste from 1950-1953; 2,550 kL (674 kgal) of P2 waste in 1964-1967; 1,180 kL (312 kgal) of BYSltCk waste in 1968; and 753 kL (199 kgal) of cesium recovery (CSR) supernatant in 1969. Based on the HDW model, this tank nominally contains 654.8 kL (173 kgal) of 2C1 solids, 41.6 kL (11 kgal) of 2C2 sludge, 458 kL (121 kgal) of 1C2 waste, 15.1 kL (4 kgal) of unknown waste, and 230 kL (61 kgal) of BYSltCk. Two push-mode cores (cores 88 and 89) were obtained from this tank in 1995 (Field et al. 1996a). Each core contained seven segments. The waste analyte profiles were found to be generally uniform and consistent, as a function of depth, for Al, Bi, Cr, Na, Si, U, Cl, F, NO_3, NO_2, PO_4, and SO_4. Consequently, these cores cannot be used to distinguish the 2C1, 2C2, 1C2, and BYSltCk layers from one another. Sub-segment data may be found in Appendix A of the TCR, but these data also support the earlier observation that distinct layers cannot be readily identified from these results. Analytical data for metals have been averaged so that only core composite results are available.

Because 50 percent of the estimated inventory consists of 2C waste, composite sample results were considered to be acceptable for the 2C tank waste layer composition. However, certain analytes were excluded, including Na, NO_3, and SO_4. These analytes were not included in the 2C tank waste layer composition because high analyte concentrations found in the sample are thought to be due to contamination from 1C waste or BYSltCk. The best estimate of the date of analysis for decay calculations is July 1, 1995.

Tank 241-B-110 received 7,960 kL (2,100 kgal) of 2C1 waste from 1945-1950; 16,450 kL (4,346 kgal) of 2C2 waste from 1950-1953; and 2,551 kL (674 kgal) of P2 waste in 1967. According to the HDW model, this tank currently contains 541 kL (143 kgal) of 2C1 waste, 378 kL (100 kgal) of 2C2 waste, and 11.4 kL (3 kgal) of P2 waste. Eight core samples (cores 1, 2, 3, 4, 9, 10, 11, and 12) were taken from this tank in 1989-1990 (Kunthara et al. 1997). Seven of these cores were analyzed (all except core 11), with only composite results being reported. Therefore, segment level data cannot be used to distinguish 2C1, 2C2, and P2 layers from one another. However, statistical analysis techniques were used to assess vertical and horizontal variability in the waste. The results show that very little variability exists in both the vertical and horizontal dimensions. Because 2C waste represents about 99 percent of the estimated inventory, composite sample results from this tank were included in the construction of the 2C waste type template. The best estimate of the date of analysis for decay calculations is April 1, 1990.

About 20,450 kL (5,402 kgal) of secondary 2C waste generated between 1945 and 1953 was transferred to tank 241-B-111. Along with this waste, tank 241-B-111 also received 1,060 kL (280 kgal) of evaporator bottoms from tank 241-B-105 in 1954; 2,530 kL (668 kgal) of P2 waste from 1964-1967; 7,063 kL (1,866 kgal) of CSR supernatant from 1969-70; and 348 kL (92 kgal) of BYSltCk concentrate from tank 241-BY-112 from 1969-1970. The HDW model indicates that the present inventory consists of 791 kL (209 kgal) of 2C2 waste, 3.78 kL (1 kgal) of decontamination waste (DW) waste, and 98.4 kL (26 kgal) of P2 waste. Two core samples (cores 29 and 30) were obtained from this tank in 1991 (Benar et al. 1997b). Each core consisted of four segments (the first segment from each core was found to be empty). In addition, composite results were also reported for each core. Based on these results, there appears to be little difference in analyte concentration as a function of depth or core profile position in the tank. This data, therefore, cannot be used to distinguish the 2C2 and P2 layers from one another. Because the 2C layer represents about 88 percent of the waste, the composite analytical results from these cores were used to estimate analyte compositions in the 2C tank waste layer. Analytical results for NO_2, however, appear to be too high because of the possible contribution from P2 waste and BYSltCk. This NO_2 value was excluded from use in the 2C waste type template. The best estimate of the date of analysis for decay calculations is February 1, 1993.

Tank 241-T-111 is known to have received 10,257 kL (2,710 kgal) of secondary 2C1 waste from 1945 to 1949; 14,038 kL (3,709 kgal) of secondary 2C2 waste in 1950-1951; 7,240 kL (1,913 kgal) secondary 2C2 and 4,618 kL (1,220 kgal) of 224 waste in 1952; and 46,975 kL (12,410 kgal) of secondary 2C2 waste from 1953 to 1956. Based on the HDW model, these transfers produced 526 kL (139 kgal) or 127.8 cm (50 in.) of 2C1 sludge, 397 kL (105 kgal) or 96.6 cm (38 in.) of combined 2C2 and 224 sludge, and a final layer containing 802.4 kL (212 kgal) or 195 cm (77 in.) of 2C2 sludge. According to the flowsheet, 224 waste should be

the only source of La and Mn, while 2C2 waste is the principal source of Ca, Fe, Si, U, Cl, CO_3, ^{137}Cs, and ^{90}Sr. Three cores (cores 31, 32 and 33) were taken from this tank in 1991 (Field et al. 1997a). Two of these cores (31 and 33) were quite successful, retrieving more than 80 percent of the waste in most segments. Each core consisted of 9 segments, representing an average sludge depth of 434.3 cm (171 in.), compared to 419 cm (165 in.) of sludge predicted by the HDW model and 410 cm (161 in.) of sludge in Hanlon reports). Based on the amount of La and Mn in 224 waste, the ratio of 224 waste to 2C waste appears to be about 25:75 (in contrast to a ratio of 8:92 predicted by the HDW model). Analytical data from segment 9 of core 31 and segments 1, 3, 5, 7, and 9 of core 33 indicate a uniform composition profile for most analytes, except Bi, Ca, La, and Mn. Segment 1 of core 33 has much lower concentrations of Bi and La, but higher concentrations of Ca and Mn than found in other segments of this core. These results are not consistent with the expected trend for 224 waste. The 224 waste should have about the same amount of Bi, but lower concentrations of Ca and higher concentrations of La and Mn than 2C waste. Because these results are consisted with the anticipated trend for 224 waste, composite results from both cores were used to represent 2C waste. However, the concentrations of La and Mn were reduced to zero to compensate for the 224 waste in this tank. The best estimate of the date of analysis for decay calculations is January 1, 1992.

According to the waste transaction records, tank 241-T-112 received 76,900 kL (20,300 kgal) of secondary 2C and 224 waste from 1945 to 1956 and 15,750 kL 4,160 kgal) of DW waste from 1960 to 1973. Based on the HDW model, this tank contains 227 kL (60 kgal) of 2C2 sludge. Two core samples (cores 185 and 186) were obtained from this tank in 1997 (McCain et al. 1998). Segment 1 from both cores did not contain any solids. A sludge slurry consistency was found in the second segments. It was decided to eliminate La and Mn from the 2C waste composition profile because these analytes were probably added with 224 waste to this tank. Aluminum, nitrite, and zirconium were also eliminated because these components are likely to be found in significant concentration only in 1C/CW waste or coating waste from an unidentified source. Finally, Pb and U were eliminated from this composition profile because of high "less than" analytical values based on fusion analysis. The best estimate of the date of analysis for decay calculations is April 1997.

The 2C waste type template composition is based on sample values from tanks 241-B-104, 241-B-110, 241-B-111, 241-T-111, and 241-T-112. The spreadsheet for the 2C waste type template composition (2C.xls) is stored on fileserver AP001\STDINV\ RPP-8847\2C.xls.

A.1.4 224 TANK WASTE LAYER ASSESSMENT

According to the HDW model waste transaction files, the lanthanum fluoride process used for plutonium purification generated "224" waste that was sent to T- and B-Farms after 1951. The *Hanford Works Monthly Report - July 1948* (GE 1948), however, reports that the 241-B-201 and 241-T-201 tanks were receiving 224 waste prior to July 1948. Initially, the 224 waste was settled in 30,000-gallon concrete settling tanks (241-361) that overflowed to dry wells and later cribs (GE 1944). After this practice for disposal of 224 waste was abandoned, the 224 waste was sent to tanks 241-B-201 and 241-T-201. Later, 224 waste was cascaded from tank 241-B-204 to tank

241-B-203 to tank 241-B-202, or from tank 241-T-204 to tank 241-T-203 to tank 241-T-202 (GE 1948). After the 200 series tanks were filled, 224 waste was also disposed of in tanks 241-T-110 and 241-T-111 (GE 1955). The tank set previously used as the basis for the 224 waste composition consisted of tanks 241-B-201, 241-B-203, 241-B-203, and 241-B-204. The HDW model waste transaction files indicate that tanks 241-T-110, 241-T-111, 241-T-201, 241-T-202, 241-T-203, and 241-T-204 also received 224 waste. Whereas the 241-T-110 and 241-T-111 tanks contain other wastes in addition to 224 waste, the B- and T-Farm 200 series tanks are believed to contain only 224 waste.

Two or more cores were taken from each of the four 241-B-200 tanks. The best estimate of the dates of analysis are September 1, 1991 for 241-B-201 and 241-B-202 (Conner et al. 1997a and LMHC 1999a). Three core samples were taken from 241-B-203 in 1995; however, only chemical analytes and total alpha were measured. Values for ^{90}Sr and ^{137}Cs are based on the average of values from 1978 and 1982 samples (Jo et al. 1996a). Core samples were taken from tank 241-B-204 in 1995 (Sasaki et al. 1996). The best-basis estimate of the date of analysis is November 1, 1995.

A single core sample was taken from each of the 241-T-200 tanks (Simpson 1998). The best estimate of the date of analysis for decay calculations is June 1, 1997.

The *Hanford Works Monthly Report - July 1948* (GE 1948) states that the first tanks to be filled with 224 wastes were 241-B-201 and 241-T-201. Internal letters also indicate that the composition of these tanks may be different from the 224 waste that was cascaded from tank 241-B-204 to tank 241-B-203 to tank 241-B-202 and from tank 241-T-204 to tank 241-T-203 to tank 241-T-202 (Bailor 1952 and Beaulieu 1952). Examination of the sample values for these tanks reveals trends for several analytes. Values associated with the 241-B-201 and 241-T-201 tanks for Bi, Fe, La, Ni, PO_4, and total alpha are noticeably higher than the values for the other 200 series tanks. Based on this observation, pre-1949 224 waste is based on the average of tanks241-B-201 and 241-T-201 values and post 1948 224 waste is based on the average of tanks 241-B-202, 241-B-203, 241-B-204, 241-T-202, 241-T-203, and 241-T-204 values.

The waste type template for 224 waste is based on sample values from tanks 241-B-201, 241-B-202, 241-B-203, 241-B-204, 241-T-201, 241-T-202, 241-T-203, and 241-T-204. The spreadsheet for the pre 1949 224 and post 1948 224 waste type template compositions (224.xls) is stored on fileserver AP001\STDINV\RPP-8847\224.xls.

A.2.0 URANIUM RECOVERY PROCESS AND SCAVENGING PROCESS WASTES

The TBP process was used to recover uranium from metal waste (MW) sluiced from the tanks. This waste is also referred to as uranium recovery or UR waste in the HDW model. Flowsheet and process history documents, however, usually refer to recovery of uranium at U-Plant as the Tributyl Phosphate Process or the TBP Process. U-Plant has also been referred to as the TBP Plant.

In-plant scavenged TBP waste is similar to TBP waste but with increased Ni and Fe content due to the use of ferrocyanide by the scavenging process. The HDW model defines two in-plant scavenged wastes, PFeCN1 and PFeCN2. The PFeCN2 waste stream is differentiated from the earlier PFeCN1 waste by reduced $Na_2NiFe(CN)_6$ content.

In-tank scavenging of TBP and 1C wastes treated only the waste supernatants. Scavenging with ferrocyanide precipitated Cs, which, after settling in the tank, allowed disposal of supernatant to cribs.

A.2.1 TBP TANK WASTE LAYER ASSESSMENT

Previously two tanks (241-BX-109 and 241-TY-105) were used either independently or in combination as a basis for the TBP waste composition. After review of the HDW waste transaction files, two additional tanks were identified with layers of TBP waste sufficiently large to be considered for inclusion in the TBP waste type template. These tanks are 241-C-108 and 241-TY-103. However, the TBP waste layers in both tanks 241-C-108 and 241-TY-103 could not be distinguished from other waste layers because of commingling and/or homogenization of the sample.

Another possible source of TBP waste is tank 241-BY-109. Although the HDW model predicts that tank 241-BY-109 contains MW and BYSltCk, Jo et al. (1998b) concluded from the tank history that tank 241-BY-109 contains TBP waste. Jo et al. (1998b) assume from sample analysis that tank 241-BY-109 contains 47.3 kL (12.5 kgal) of CWP, 168 kL (44.5 kgal) of TBP, and 61.6 kL (233 kgal) of BYSltCk with CWP on the bottom. The CWP layer was assumed due to a high Al concentration in the lowest half core segment. Although this layer may be cladding waste, PUREX as the source is unlikely because CWP waste was not generated before 1956 and the TBP was added to the tank during the second quarter of 1956. Based on review of the core profiles, only segment 7 of core 203 would have intercepted the TBP waste layer. Review of the core segment data did not reveal an identifiable TBP waste layer. This may be due to commingling of the waste within the tank or the fact that the TBP waste receipt may only have consisted of supernatant.

The HDW model predicts that tank 241-BX-109 contains 129 kL (34 kgal) of 1C2 waste under a 602 kL (159 kgal) layer of TBP waste. Because the bottom of the tank is dished and the core samples were taken near the outer edge of the tank, 1C2 waste should only be apparent in the lower half of segment 4. Review of the segment data did not reveal this layer. A high aluminum concentration in the top segment indicates that cladding waste supernatant may have commingled with the TBP waste. However, because no other layer indicators were found, the composition of tank 241-BX-109 is assumed to be essentially all TBP waste (Field et al. 1996b). The best estimate of the date of analysis for decay calculations is May 1, 1995.

The HDW model waste transaction files indicate that tank 241-TY-105 contains 598 kL (158 kgal) of TBP waste and 276 kL (73 kgal) of an unknown saltcake. Because the core sample of this tank recovered only material from the top 94 cm (37 in.), the validity of the sample as TBP waste could be questioned (Weiss and Mauss 1987d). An alternate source of information (Anderson 1990) indicates that the tank is all TBP waste. Based on the similarity in waste composition between the waste in tank 241-TY-105 and tank 241-BX-109, the waste in tank 241-TY-105 is considered to be all TBP waste. The concentrations of Al, Fe, and Cr are considered to be key indications of TBP waste. The best estimate of the date of analysis for decay calculations is October 1, 1985.

The composition of the TBP waste type template is based on sample values from tanks 241-BX-109 and 241-TY-105. The spreadsheet for the TBP waste type template composition (TBP.xls) is stored on fileserver AP001\STDINV\RPP-8847\TBP.xls.

A.2.2 1CFeCN TANK WASTE LAYER ASSESSMENT

The 1CFeCN waste tank layer is a ferrocyanide sludge that resulted from in-farm scavenging of T-Plant 1C supernatant in 241-TY Farm to remove ^{137}Cs by precipitation with $Na_2NiFe(CN)_6$. Only three tanks (241-TY-101, 241-TY-103, and 241-TY-104) are believed to have received 1CFeCN waste.

Based on Agnew et al. (1997a), tank 241-TY-101 should contain approximately 174 kL (46 kgal) of T1SltCk under 273 kL (72 kgal) of 1CFeCN. Review of the core recovery data indicates that the core sample retrieved mostly T1SltCk and little 1CFeCN waste (Weiss and Mauss 1987a).

The HDW model waste transaction files indicate that less than 30 percent of the waste in tank 241-TY-103 is 1CFeCN waste. Although multiple core samples were taken from this tank in 1985, the samples were composited before analysis and, therefore, are not usable for this assessment (Weiss and Mauss 1987b).

Tank 241-TY-104 is believed to contain two-thirds 1CFeCN waste and one-third TBP waste. The HDW model assumes that the TBP waste is on top while Anderson (1990) indicates that the TBP entered the tank first. This analysis assumes that the TBP is on the bottom of the tank. Six core samples were recovered from tank 241-TY-104 in 1985 and combined into a composite sample (Weiss and Mauss 1987c). Because of the dished bottom of the tank and the location of sampled risers, it is reasonable to conclude that the composite contains little TBP waste and is mostly 1CFeCN waste. The best estimate of the date of analysis for decay calculations is September 1, 1985.

The composition of the 1CFeCN waste type template is based only on sample values from tank 241-TY-104. The previous assessment of 1CFeCN waste was also based only on tank 241-TY-104. However, the concentration values were obtained by a simple averaging of core sample results. Because two of the core samples came from the same riser, this method could introduce a bias to the values. The composition of 1CFeCN waste shown in this report uses a

weighted averaging to compute the concentration values. The spreadsheet for the 1CFeCN waste type template composition (1CFeCN.xls) is stored on fileserver AP001\STDINV\RPP-8847\1CFeCN.xls.

A.2.3 PFeCN TANK WASTE LAYER ASSESSMENT

The PFeCN waste type consists of TBP waste treated in-plant with $Na_2NiFe(CN)_6$. The HDW model (and process flowsheets) indicate that two PFeCN waste types, PFeCN1 and PFeCN2, were generated. PFeCN1, generated during 1954 and 1955, has approximately twice the nickel ferrocyanide content of the PFeCN2 that was generated between 1956 and 1958. The seven tanks identified by the HDW model as having received PFeCN1 or PFeCN2 are tanks 241-BY-104, 241-BY-105, 241-BY-106, 241-BY-107, 241-BY-108, 241-BY-110, and 241-BY-112. The previous PFeCN template was based only on waste samples from tanks 241-BY-106 and 241-BY-110. Although PFeCN can be readily distinguished from other waste types based on Bi, Ca, Fe, Ni, and P content, PFeCN1 and PFeCN2 proved to be indistinguishable from each other. A contributing factor may have been that much less PFeCN type waste was found in the tanks than was predicted by the HDW model.

Based on the HDW model waste transaction files, tank 241-BY-104 is believed to contain 568 kL (150 kgal) of PFeCN2 under 674 kL (178 kgal) of BYSltCk. The core profile indicates that core sampling recovered waste to within 13 cm (5 in.) of the lowest point in the tank bottom. Therefore, only the bottom two and one-half segments would be expected to be PFeCN2 waste. Two cores were collected from this tank. Elevated levels for Bi, Ca, Fe, Pb, Ni, Si, Sr, P, U, and ^{90}Sr are found in the three lowest sub-segments of core 116 (7D, 8A and 8D). This composition (and the location) appears consistent with a ferrocyanide waste layer. The lowest segments of core 117 did not exhibit any elevated Ni levels and is not believed to represent ferrocyanide waste (Benar et al. 1996). Core segments 116:7D, 116:8A, and 116:8D define the PFeCN2 waste layer composition in tank 241-BY-104. The best estimate of the date of analysis for decay calculations is December 1, 1995.

Tank 241-BY-105 is presumed to contain 538 kL (142 kgal) PFeCN2 under 1,276 kL (337 kgal) of BYSltCk. The tank also contains 57 MT (63 tons) of Portland cement. Review of the core profile data indicates that the sampling activity did not intercept the ferrocyanide waste layers (Simpson et al. 1996b).

Tank 241-BY-106 is believed to contain 204 kL (54 kgal) of PFeCN1 under 167 kL (44 kgal) PFeCN2 and 2,059 kL (544 kgal) BYSltCk. Core 121 intercepted only the saltcake layer. Cores 64 and 65 were taken from the same riser near the perimeter of the tank. The dished bottom of the tank makes it unlikely that the bottom 23 to 45 kL (6 to 12 kgal) of waste were intercepted by the core sample. Although the entire bottom segment of cores 64 and 65 was expected to be ferrocyanide sludge, core 64 contained only saltcake and the bottom segment of core 65 contained both saltcake and sludge. Owing to the small amount of sludge sampled, it was not possible to establish if the layer was PFeCN1 or PFeCN2 (Bell et al. 1996). Only core segment 65:13U is included in the PFeCN waste type template. The best estimate of the date of analysis for decay calculations is February 1, 1995.

The HDW model assumes the following waste type layers exist in tank 241-BY-107: 91 kL (24 kgal) 2C2, 76 kL (20 kgal) TBP, 163 kL (43 kgal) of PFECN1, 114 kL (30 kgal) of PFeCN2, and 564 kL (149 kgal) of BYSltCk. During sampling, only four of the expected six core segments were taken before the tank floor was encountered. Only the lowest half segments of cores 151 and 161 contain sludge. Visual inspection and segment composition indicate the sludge is PFeCN waste (McCain et al. 1997). Based on the core profiles constructed for this tank, tank 241-BY-107 contains 98 kL (26 kgal) of sludge under 693 kL (183 kgal) of saltcake. The discrepancies with the volume sampled between the process history and the tank waste volume make it unclear as to whether PFeCN1 or PFeCN2 was sampled in tank 241-BY-107. Core segments 151:4L, and 161:4L are believed to be PFeCN waste due to the presence of Fe (27,000 and 40,000 μg/g) and more importantly a high concentration of Sr (8,900 to 12,000 μg/g). These core segments define the PFeCN layer in tank 241-BY-107. The best estimate of the date of analysis for decay calculations is September 1, 1996.

The HDW model waste transaction files indicate that tank 241-BY-108 contains 503 kL (133 kgal) of PFeCN1, 121 kL (32 kgal) of PFeCN2, and 201 kL (53 kgal) of BYSltCk. Hanlon (1999) indicates that the tank contains 583 kL (154 kgal) sludge and 280 kL (74 kgal) of saltcake. From examination of the core sample chemical analysis, only the bottom segment (197 kL [52 kgal]) exhibits the chemical characteristics of PFeCN. The segment just above the bottom segment appears to contain a mix of PFeCN waste and other wastes (Baldwin et al. 1996a). Core segments 99:4A, 99:4B, 99:4C, 99:4D, 104:5B, 104:5C, and 104:5D define the PFeCN layer in tank 241-BY-108. The best estimate of the date of analysis for decay calculations is September 1, 1995.

The HDW model waste transaction files indicate that tank 241-BY-110 contains 140 kL (37 kgal) 1C2, 420 kL (111 kgal) PFeCN1, 159 kL (42 kgal) PFeCN2, and 787 kL (208 kgal) BYSltCk. Hanlon (1999) indicates that this tank contains 390 kL (103 kgal) sludge and 1,117 kL (295 kgal) saltcake. From examination of the core sample chemical analysis, an uneven sludge distribution exists in the tank. All of the sludge intercepted appears to be PFeCN. There was no evidence of 1C waste in the core samples (Schreiber et al. 1996). Core segments 101:8C, 101:8D, 101:9A-9D, 103:8D, 103:9A-9D, 107:8A, 107:8B, 107:9A-9D, 113:7A-7D, and 113:8A-8D define the PFeCN layer in tank 241-BY-110. The best estimate of the date of analysis for decay calculations is December 1, 1995.

The HDW model indicates that only a few thousand gallons of PFeCN2 were sent to tank 241-BY-112. There is no evidence from the core samples that a PFeCN2 layer exists in this tank (Baldwin et al. 1997a).

The composition of the PFeCN waste type template is based on the sample values from tanks 241-BY-104, 241-BY-106, 241-BY-107, 241-BY-108, and 241-BY-110. The spreadsheet for the PFeCN waste type template composition (PFeCN.xls) is stored on fileserver AP001\STDINV\RPP-8847\PFeCN.xls.

A.2.4 TFeCN TANK WASTE LAYER ASSESSMENT

The TFeCN waste type is a ferrocyanide sludge that resulted from in-farm scavenging of TBP Process supernatant in CR Vault. Only four tanks (241-C-108, 241-C-109, 241-C-111, and 241-C-112) are known to have received TFeCN waste. The previous waste type template was based on a sample from only tank 241-C-109.

The TFeCN layer in tank 241-C-108 is less than 13 cm (5 in.) thick and is layered between other waste types. Also, an auger was used to sample the tank (Place 1998d). Auger samples are more difficult to correlate to specific layers. These factors rule out use of tank 241-C-108 samples to establish TFeCN waste composition.

Tank 241-C-109 is presumed to contain 170 kL (45 kgal) of TFeCN between 37.9 kL (10 kgal) 1C1 and 26.5 kL (7 kgal) of hot semi-works (HS) waste. Single segment core samples were taken each from three different risers. The tank has a dished bottom, and the three risers are near the perimeter of the tank (Simpson et al. 1997a). These factors may have minimized the amount of 1C1 waste intercepted by the core samples. The core samples may have a high ^{90}Sr bias due to the HS waste. The waste also exhibits a high Al content. Aluminum is not considered to be native to the TFeCN waste type, and the composition was adjusted accordingly. The best estimate of the date of analysis for decay calculations is October 1, 1992.

The HDW waste transaction files indicate that tank 241-C-111 contains only a 5 cm (2 in.) layer of TFeCN waste between layers of 1C1 and CWP1. This small layer would not be distinguishable from the adjacent layers. Three core samples were obtained from this tank. However, recovery was poor, and core profile information is limited (Benar et al. 1997c)

Tank 241-C-112 is reported by the HDW transaction files to contain, from the bottom up, 57 kL (15 kgal) 1C1, 257 kL (68 kgal) TFeCN, 49 kL (13 kgal) CWP1, 11.4 kL (3 kgal) CWP2, 3.79 kL (1 kgal) HS, and 15.1 kL (4 kgal) TFeCN. Three two-segment cores were taken from three widely spaced risers. The sample data exhibit large concentrations of Fe and Ni. The riser locations and dished bottom of the tank should have minimized the amount of 1C1 waste intercepted by the core samples (Simpson et al. 1997b). The best estimate of the date of analysis for decay calculations is April 1, 1992.

Both tanks 241-C-109 and 241-C-112 are expected to be better than 70 vol% TFeCN waste. The composition of the TFeCN waste type template is based on the composite sample values from tanks 241-C-109 and 241-C-112. The spreadsheet for the TFeCN waste type template composition (TFeCN.xls) is stored on fileserver AP001\STDINV\RPP-8847\TFeCN.xls.

A.3.0 REDOX PROCESS WASTES

The REDOX process generated several different waste types: cladding waste including CWR1 and CWR2, high-level waste including R1 and R2, and a high salt waste, RSltCk. The REDOX cladding waste from aluminum clad fuel is divided by the HDW model into two types, CWR1 and CWR2. The CWR1 was generated by REDOX from 1952 to 1960, and CWR2 was generated from 1961 to 1967. A high aluminum content is characteristic of REDOX cladding waste. CWR2 has a higher Al/Na ratio than does CWR1. The REDOX cladding waste was frequently intermixed with REDOX high-level waste in the same storage tanks. The HDW model divides the high-level waste into two campaigns; R1 is from 1952 to 1958 and R2 is from 1959 to 1966. R1 and R2 also have high aluminum concentrations. R2, because of changes to the process flowsheet, has a higher Al/Na ratio than R1. REDOX high-level waste is expected to have more iron and nickel than the cladding waste. Also, REDOX high-level waste exhibits a much higher ^{90}Sr concentration than does cladding waste. The ^{90}Sr concentration is a considered to be a key analyte for differentiating REDOX high-level waste from cladding waste.

Waste type templates were only developed for the R1 and CWR1 waste types. The tanks that contain the R2 and CWR2 waste types are far fewer in number than the tanks that contain R1 and CWR1 waste types. For those tanks that received R2 and CWR2 wastes, core segments containing these waste often appeared to be contaminated from adjacent waste layers. Radiolytic decay heat from REDOX high-level waste caused wastes in many of the REDOX high-level waste receiver tanks in 241-S and 241-SX Farm to boil at atmospheric pressure. This concentration process produced the RSltCk waste found in many of these tanks. Although there is some uncertainty with respect to which tanks experienced boiling conditions, the RSltCk layers are expected to be distinguishable from the R1 and R2 wastes by higher Na concentrations and lower ^{90}Sr concentrations in the RSltCk layers.

A.3.1 CWR1 TANK WASTE LAYER ASSESSMENT

The tanks previously used as the basis for the CWR1 waste type template consisted of tanks 241-S-104, 241-S-107, and 241-U-204. Review of the process history documents indicates that tank 241-S-104 is more likely to contain R1 than CWR1. Review of the process knowledge assessments published in the BBI documents indicates that of the sampled tanks, 241-S-107, 241-U-109, 241-U-110, and 241-U-204 were the most likely to have identifiable layers of CWR1 waste.

Anderson (1990) indicates that 241-S-107 contains about 180 cm (6 ft) of CWR1 waste under 140 cm (4.5 ft) of evaporator bottoms and saltcake. The HDW model waste transaction files indicate that tank 241-S-107 contains a more complex inventory. The HDW model assumes 86 cm (34 in.) of R1 waste under 227 cm (60 in.) of cladding waste and 102 cm (40 in.) of saltcake. The TCR (Simpson et al. 1996a) includes a statistical cluster analysis based on 13 waste constituents. The cluster analysis did not distinguish the CWR1 layer from the R1 layer. However, based on ^{90}Sr, Ni, Pb, and Zr, the R1 and CWR1 layers can be distinguished

from each other. A high ^{90}Sr concentration is a key indication of R1 waste relative to CWR1. The high ^{90}Sr concentrations in core segments 110:7, 110:8, 111:7, 111:8, 105:7L, and 105:8 correlate well with the expected location for the R1 waste layer, whereas the higher Ni and Pb values found above the 110:5, 111:6 and 105:6 segments are indicators of cladding waste generated in the late 1950's. These indicators identify the CWR1 layer as core 105, segments 6 and 7U; core 110, segments 5 and 6; and core 111, segment 6 (Simpson et al. 1996a). The best estimate of the date of analysis of this sample for decay purposes is October 1, 1995.

The engineering assessment for the BBI concludes that there are 132 kL (35 kgal) of CWR1 on the bottom of tank 241-U-109. The overlying wastes are S1SltCk and S2SltCk. Only the core segments nearest the tank bottom are expected to have intercepted the CWR1 waste layer. These segments are core 123, segments 9B and C; core 124, segment 9L; and core 128, segment 9. However, the low aluminum concentration and high phosphate concentration of the 124:9L segment indicate that this sample is not CWR1 but more likely is saltcake (Baldwin and Stephans 1996). The best estimate of the date of analysis of this sample for decay purposes is February 1, 1996.

The HDW model predicts that tank 241-U-110 contains only MW and 1C1. The tank is also known to have received R1 and CWR1 wastes (Agnew et al. 1997b). The conclusion of the BBI assessment is that core segments 7:1, 8:1, and 14:1 contain CWR1. This assessment is based on the high Al concentration, which is characteristic of CWR1 and R1 waste and the absence of Bi which is typically found in 1C1 waste (Bell and Stephans 1997). The best estimate of the date of analysis of this sample for decay calculations is January 1, 1990.

The HDW model waste transaction files indicate that tank 241-U-204 received only diversion box flush water and CWR1 waste. Anderson (1990) classifies the flush water as diluted REDOX high-level waste and Bismuth Phosphate process second cycle decontamination waste. The CWR1 is believed to have been transferred from tank 241-U-110. Two cores samples (cores 81 and 82) taken in 1995 have been analyzed by inductively coupled plasma (ICP) (Raphael and Tran 1995). This data was supplemented by sample results from a 1978 core sample. Values for Cl, F, K, NO_2, NO_3, ^{90}Sr, and ^{137}Cs are from the 1978 core sample. The assumed date of analysis of the 1978 sample for decay calculations is December 1, 1978.

The composition for the CWR1 waste type template is based on sample values from tanks 241-S-107, 241-U-109, 241-U-110, and 241-U-204. The spreadsheet for the CWR1 waste type template composition (CWR1.xls) is stored on fileserver AP001\STDINV\RPP-8847\CWR1.xls.

A.3.2 R1 TANK WASTE LAYER ASSESSMENT

The samples previously used as the basis for the R1 waste type template were from tanks 241-S-101, 241-S-104, and 241-S-107. Review of the process knowledge assessments contained in the BBIs indicates that of the tanks which have been sampled, tanks 241-S-111, 241-SX-101, 241-SX-103, 241-SX-105, 241-SX-108, and 241-U-103 are likely to have identifiable layers of R1 type waste. The HDW model waste transaction files (Agnew et al. 1997a) indicate that tank

241-S-104 contains about 500 kL (132 kgal) of mixed R1 and CWR1 under 609 kL (161 kgal) of RSltCk. The R1/CWR1 mix is approximately 80 percent R1 waste. Anderson (1990) indicates that all of the waste is REDOX high-level waste. Few samples were analyzed on a segment basis; most of the samples were combined into composite samples. Because of the lack of segment data, the R1 waste in tank 241-S-104 cannot be identified as a separate layer from the RSltCk waste.

The HDW model waste transaction files indicate that tank 241-S-101 contains approximately 76 cm (30 in.) of R1 waste under saltcake. The layers above the R1 layer consist of several thin layers of R1 and CWR1, RSltCk, S1SltCk, and S2SltCk. Segments 5-9 of cores 138 and 142 exhibit a high aluminum content. The only distinguishing characteristic was a decrease in phosphate below segment 7. The phosphate decrease is smeared across several segments and is not abrupt. The R1 waste would be expected to be higher in ^{90}Sr than the overlying layers; however, ^{90}Sr segment data were not available for examination (Kruger et al. 1996). Based on the R1 waste depth assumed by the HDW model, the R1 layer should be intercepted by segments 138:8L, 138:9U, 138:9L, 142:7L, 142:8U, and 142:8L. These segments all exhibit decreased phosphate concentrations. The best estimate of the date of analysis of this sample for decay purposes is May 1, 1996.

Anderson (1990) indicates that tank 241-S-107 contains about 180 cm (6 ft) of CWR1 waste under 140 cm (4.5 ft) of evaporator bottoms and saltcake. The HDW model waste transaction files indicate that tank 241-S-107 contains a more complex inventory. The HDW model assumes 86 cm (34 in.) of R1 waste under 152 cm (60 in.) of cladding waste and 102 cm (40 in.) of saltcake. The TCR (Simpson et al. 1996a) includes a statistical cluster analysis based on 13 waste constituents. The cluster analysis did not distinguish the CWR1 layer from the R1 layer. However, based on ^{90}Sr, Ni, Pb, and Zr, the R1 and CWR1 layers can be distinguished from each other. A high ^{90}Sr concentration is a key indication of R1 waste relative to CWR1. The high ^{90}Sr concentrations in core segments 110:7, 110:8, 111:7, 111:8, 105:7L, and 105:8 correlate well to the expected location for R1, whereas the higher Ni and Pb values found above 110:5, 111:6 and 105:6 segments are indicators of cladding waste generated in the late 1950's. These indicators define the R1 layer as core 105, segments 7L and 8; core 110, segments 7 and 8; and core 111, segment 7 and 8. The best estimate of the date of analysis of this sample for decay purposes is October 1, 1995.

The HDW model predicts that tank 241-S-111 contains 250 kL (66 kgal) of R1 high-level waste, 45 kL (12 kgal) of CWR1, 231 kL (61 kgal) of an unknown waste, and 1,510 kL (399 kgal) of saltcake (S1SltCk). Except for the S1SltCk, all of waste was received before 1957. Only core 149 intercepted the R1, CWR1, and the unknown (UNK) waste layers (Conner et al. 1997b). Based on the HDW model waste volumes, only segment 11 should intercept the R1 layer. However, review of the chemical and radiochemical values indicate that segments 10 and 11 are very similar, whereas the waste above segment 10 is distinctly different from the waste intercepted by segments 10 and 11. Both segments 10 and 11 are considered to be R1 because the volume of CWR1 estimated by the HDW model would account for less than a quarter of a core segment. The assumed date of analysis of this sample for decay purposes is July 1, 1996.

The HDW model predicts that tank 241-SX-101 contains 583 kL (154 kgal) of R1 waste, under 591 kL (156 kgal) of RSltCk, 352 kL (93 kgal) of S1SltCk, and 197 kL (52 kgal) of S2SltCk. Based on physical appearance, the BBI concludes that the tank contains 485 kL (128 kgal) of sludge like solids over 1,188 kL (314 kgal) of saltcake-like material on the bottom. Hu et al. (1998) concluded that the tank contains all saltcake because metals typically found in sludge were present in only low concentrations. Review of the core profile indicates that only core segments 225:6, 225:7, 225:8, and 227:7 would be expected to intercept the HDW model predicted R1 waste layer. These segments exhibit an increase in Fe and Mn content, relative to the rest of the tank, which indicate the presence of R1 sludge. However, the primary distinguishing characteristic of R1 waste, a high ^{90}Sr concentration cannot be established because ^{90}Sr segment data are not available for this tank. When compared to the other candidate tanks (241-S-101, 241-S-107, 241-S-111, 241-SX-103, and 241-SX-105), the Al concentration is less than half the mean concentration and the Na concentration is more than twice the mean concentration in these tanks. Based on these observations, it is believed that the R1 waste in this tank is commingled with saltcake and is not suitable for inclusion in the R1 waste type template.

The HDW model predicts that tank 241-SX-103 contains 37.8 kL (10 kgal) of R1 waste, 386 kL (102 kgal) of RSltCk, and 2,040 kL (539 kgal) of S1SltCk. Two cores samples were removed from the tank in April and May 1998 (LMHC 1999b). The core profile and the segment data indicate that the tank contains 150 to 380 kL (40 to 100 kgal) of sludge. Based on the presence of Al, Mn, Sr, and ^{90}Sr, segments 235:11L, 235:12AL, 239:RU, 239:11L, and 239:12L are considered to be R1 waste. The best estimate of the date of analysis for decay calculations is June 1, 1998.

The HDW waste transaction files show that tank 241-SX-105 contains 56.8 kL (15 kgal) of R1 waste, 117 kL (31 kgal) of RSltCk, 34 kL (9 kgal) of R2 waste, 68 kL (18 kgal) T2SltCk, and 2,310 kL (610 kgal) of S1SltCk. Two core samples were removed from the tank in March and May 1998 (LMHC 1999c). The segment data indicate that the lower core segments (229:12AL, 229:13U, 229:13L, 233:12L, and 233:13AL) contain sludge. These core segments correspond to a sludge height of approximately 68 cm (27 in.), or about 380 kL (74 kgal) of sludge. The segments were identified by a nested analysis of variance (ANOVA) model and by inspection. All of these segments exhibit elevated Al content, and segments 229:13U and 22913L have increased Fe and Mn concentrations relative to other waste in the tank. The Sr content is not distinctive from the rest of the tank, and ^{90}Sr was not analyzed. The lower core segments (229:12AL, 229:13U, 229:13L, 233:12L, and 233:13AL) are considered to be R1 waste. The best estimate of the date of analysis for decay calculations is June 1, 1998.

The HDW model predicts that tank 241-SX-108 contains 261 kL (69 kgal) of R1 waste and 68 kL (18 kgal) of R2 waste. The tank was sampled with an auger in September 1995 (Eggers et al. 1996a). The auger sample was analyzed by the 222-S Laboratory as upper and lower halves. There was no discernable difference between the halves. An independent analysis of the sample was made by Pacific Northwest National Laboratory (PNNL). Whereas the 222-S Laboratory sample was digested in acid, the PNNL sample was prepared by KOH and sodium peroxide fusion. Density was not determined for any of the samples. A density value of 1.75 g/mL is predicted by the HDW model. The best estimate of the date of analysis for decay calculations is

October 1, 1995. Several of the analytical values (Al, Bi, Fe, La, Na, ^{90}Sr, and total alpha) in the tank 241-SX-108 sample were observed to differ significantly from the balance of the tanks identified as R1 waste. Because of these differences, tank 241-SX-108 is excluded from the R1 waste type template.

The HDW model predicts that tank 241-U-103 contains 121 kL (32 kgal) of MW, 1,143 kL (302 kgal) of S1SltCk, and 456 kL (121 kgal) of S2SltCk. The BBI assessment concludes that the MW was removed and that the sludge layer is R1 waste (Sasaki et al. 1998). Only the lowest half segment (segment 10L) of core 182 would be expected to intercept the R1 layer. However, the 10L segment of core 182 (and other segments) exhibited relatively high PO_4 values. This implies that this waste layer is either MW, or R1 waste adulterated by MW. Because of this and other anomalies, the sample from tank 241-U-103 is considered to be nonrepresentative of R1 waste.

The composition for the R1 waste type template is based on sample values from tanks 241-S-101, 241-S-107, 241-S-111, 241-SX-103, and 241-SX-105. The spreadsheet for the R1 waste type template composition (R1.xls) is stored on fileserver AP001\STDINV\RPP-8847\R1.xls.

A.3.3 RSltCk TANK WASTE LAYER ASSESSMENT

A waste type template for RSltCk layer was not previously developed. The RSltCk was generated by the self-concentration of supernatants in the 241-S and 241-SX Farms by the radiolytic decay heat from the associated REDOX high-level waste. Although there is some uncertainty in the process records with respect to which tanks reached boiling conditions, the RSltCk layers are distinguishable from the R1 and R2 wastes by the Na and ^{90}Sr concentrations. The Na concentrations in RSltCk are expected to be higher than in the R1 and R2 waste layers and the ^{90}Sr concentrations are expected to be lower. Also, RSltCk is expected to contain more Al than S1SltCk or S2SltCk. During operation of the 242-S Evaporator, the Al-rich liquor was decanted from the saltcake solids; however, during self-concentration of the RSltCk, there was little or no removal of the Al-rich material. Of the fifteen 241-S and 241-SX Farm tanks cited by Agnew et al. (1997a) as having accumulated RSltCk waste, seven have been sampled. The seven tanks are 241-S-101, 241-S-104, 241-S-107, 241-SX-101, 241-SX-103, 241-SX-105, and 241-SX-106. With the exception of tank 241-S-104, all of these tanks also received S1SltCk and S2SltCk wastes.

Based on the HDW model waste transaction files, the RSltCk layer in tank 241-S-101 is about 80 cm (31 in.) thick and lies between the R1 waste and S1SltCk layers. The RSltCk is expected to be found in segments 5 and 6 of cores 138 and 142. Segments 5 through 9 exhibit a high aluminum content. A decrease in phosphate is observed below segment 7. Segments 1 through 4 were identified as being S1SltCk and S2SltCk based on position and aluminum content. Segments 7L and lower are believed to be R1 waste based on position and phosphate content (Kruger et al. 1996). Segments 5 and 6 of cores 138 and 42 approximate the position expected for RSltCk. Smearing between the layers is evident. Segments 138:6U, 138:6L, 138:7U,

142:5L, 142:6U, and 142:6L are thought to be the best examples of RSltCk in tank 241-S-101 based on core profile trends for Cr, K, Sr, and NO_3. The best estimate of the date of analysis for decay calculation is May 1, 1996.

The HDW model waste transaction files indicate that tank 241-S-104 contains about 500 kL (132 kgal) of mixed R1 and CWR1 wastes under 609 kL (161 kgal) of RSltCk. Anderson (1990) indicates that all of the waste is REDOX high-level waste. Few samples were analyzed on a segment basis; most of the samples were combined into composite samples. Because of the lack of segment data the, RSltCk layer, if it exists, is indeterminate from the R1 waste.

The HDW model indicates that tank 241-S-107 contains a 12 cm (4.7 in.) thick layer of RSltCk between layers of R1 and cladding waste. This layer is much too thin to be accurately identified from the core segment data.

The HDW model predicts that tank 241-SX-101 contains 583 kL (154 kgal) of R1, under 591 kL (156 kgal) of RSltCk, 352 kL (93 kgal) of S1SltCk, and 197 kL (52 kgal) of S2SltCk. Review of the core profile indicates that only core segments 225:3, 225:4, 225:5, 227:4, 227:5, and 227:6 would be expected to intercept the RSltCk layer. The segments below this layer exhibit increased Sr, Mn, and Fe, indicating R1 sludge. However, the high Na content of the bottom layer indicates that the sludge is mixed with saltcake. Sub-segment 225:4L was excluded because of apparent commingling with sludge from an unknown source. The best estimate of the date of analysis for decay calculations is February 1, 1998.

The HDW model predicts that tank 241-SX-103 contains 37.9 kL (10 kgal) of R1 waste, 386 kL (102 kgal) of RSltCk, and 2,040 kL (539 kgal) of S1SltCk. Two cores samples were removed from the tank in April and May 1998 (LMHC 1999b). The core profile and the segment data indicate that the tank contains 150 to 380 kL (40 to 100 kgal) of sludge. The latter volume appears more probable. Based on the presence of Al, Mn, Sr, and ^{90}Sr, segments 235:11L, 235:12AL, 239:RU, 239:11L, and 239:12L are considered to be R1 waste. Anderson (1990) indicates that there were 223 kL (59 kgal) of solids present in the tank prior to receipt of S1SltCk waste. This volume reconciles well with the sludge volume determined from the core profiles. The low Al content of the saltcake layer confirms that the tank does not contain any RSltCk.

The HDW model waste transaction files indicate that tank 241-SX-105 contains 56.7 kL (15 kgal) of R1 waste, 117 kL (31 kgal) of RSltCk, 34 kL (9 kgal) of R2 waste, 68 kL (18 kgal) of T2SltCk, and 2,309 kL (610 kgal) of S1SltCk. The RSltCk layer is only 29 cm (11.3 in.) thick. This layer is much too thin to be accurately identified from the core segment data.

According to the HDW model waste transaction files, tank 241-SX-106 contains only 3.79 kL (1 kgal) of RSltCk on the very bottom of the tank. This layer is too thin to be accurately identified from the core segment data and is unlikely to have been intercepted by the core sampler.

The composition for the RSltCk waste type template is based on sample values from tanks 241-S-101 and 241-SX-101. The spreadsheet for the RSltCk waste type template composition (RSltCk.xls) is stored on fileserver AP001\STDINV\RPP-8847\RSltCk.xls.

A.4.0 PUREX PROCESS WASTES

The PUREX process generated several different compositions of high level waste and cladding waste streams due to process improvements and evolution of reactor fuel design.

The HDW model subdivides the PUREX high-level waste into three different campaigns, P1, P2, and P3. The P1 waste was generated by the PUREX process from 1956 to 1962; P2 waste was generated from 1963 to 1967; P2' waste from 1968 to 1972; and P3 waste from 1983 to 1988. The HDW model waste transaction files indicate that P1 waste was sent to only two tanks, 241-A-101 and 241-A-104. The change from P1 waste to P2 waste coincides with addition of a second cycle uranium extraction recycle and sugar denitration of the high-level waste to the PUREX process flowsheet, thereby reducing the nitric acid content of the waste and subsequent caustic addition. The P2' waste was sent directly to B-Plant without caustic addition for strontium removal. Subsequent to processing at B-Plant, the waste was sent to the underground tanks for storage and is tracked by the HDW model as B-Plant high-level (B) and low-level (BL) wastes. The P3 waste was transferred directly to the DST aging waste tanks, 241-AZ-101 and 241-AZ-102. Both of the 241-AZ tanks have been sampled. Waste from the P2 era is the only fraction of PUREX high-level waste dispersed over multiple tanks. Much of the P2 waste was sluiced for strontium recovery in B-Plant, and the neutralized waste was returned to SSTs. The HDW waste transaction files indicate layers of P2 waste reside in eight SSTs. Review of the process knowledge assessments contained in the BBI documents indicate that of the sampled tanks, 241-A-105, 241-AX-103, 241-AX-104, and 241-B-111 are likely to have identifiable layers of P2 waste.

PUREX cladding waste from aluminum clad fuel is divided by the HDW model into two periods, CWP1 and CWP2. The CWP1 waste was generated by PUREX from 1956 to 1960 while CWP2 waste was generated from 1961 to 1972. Cladding waste is characterized by a high aluminum content. CWP2 has a higher Al/Na ratio than CWP1 waste. CWP2 is more likely to be contaminated with Th, Zr, and SRR than is CWP1 waste.

A.4.1 P2 TANK WASTE LAYER ASSESSMENT

The waste in tank 241-A-105 consists of residual P2 waste that was not removed by sluicing for Sr recovery. During sluicing, the sludge was periodically sprinkled with inhibited sulfuric acid followed by neutralization. Sluicing was stopped on November 17, 1970, and a sludge sample was obtained in 1972 (Lambert 1998). The assumed date of analysis of this sample for decay purposes is January 1, 1972.

Tank 241- AX-103 received P2 waste and organic wash waste (OWW3) prior to being sluiced in 1976. Saltcake (A1SltCk) was added over the residual sludge layer in 1980. A core sample was taken and analyzed in 1997. This sample did not intercept the P2 layer. A sludge sample was also taken in 1974 and analyzed in 1976 (Jo et al. 1998a). The measured sludge density of 2.03 g/mL was used to convert the 1974 sample results to μg/g units. Although this sludge sample was taken before sluicing, sluicing is unlikely to have significantly changed the composition. Sluicing fluids were generally saturated supernatants, and much of the residual material would be sludge that did not dislodge. A Sr value was determined from the 1974 sample analysis. The assumed date of analysis of this sample for decay purposes is December 1, 1976.

Tank 241-AX-104 accumulated P2 sludge from the third quarter of 1966 until the second quarter of 1969. The tank was sluiced for strontium recovery during 1977 and 1978 and did not receive additional waste. A grab sample of the sludge heel was obtained in 1997 (Field et al. 1998a). The reported approximate date of ^{90}Sr and gamma energy analysis (GEA) analysis was March 1, 1998.

The HDW waste transaction files indicate that tank 241-B-111 contains 98.4 kL (26 kgal) of P2 waste on top of 2C2 waste. Two core samples were taken from tank 241-B-111, but the core samples were homogenized, and data from the P2 layer is not available (Benar et al. 1997b).

The composition for the P2 waste type template is based on sample values from tanks 241-A-105, 241-AX-103, and 241-AX-104. The spreadsheet for the P2 waste type template composition (P2.xls) is stored on fileserver AP001\STDINV\RPP-8847\P2.xls.

A.4.2 CWP1 TANK WASTE LAYER ASSESSMENT

The previous basis for CWP1 waste type template consisted of only one tank, 241-C-105. Review of Agnew et al. (1997a) and tank sampling records indicates that tanks 241-C-103, 241-C-104, and 241-C-111 also contain sizable amounts of CWP1 waste.

The CWP1 sludge in tank 241-C-103 is on the bottom of the tank under a layer of washed PUREX sludge (AR) solids (Place 1998a). Segment 4 of core 63 is believed to have intercepted the CWP1 layer. The sample exhibits a high Al value and a moderate ^{90}Sr value. The reported nickel value is not reliable as a sludge identifier because the sample fusion was conducted in a Ni crucible. The assumed date of analysis of this sample for decay purposes is February 1, 1995.

According to the HDW model waste transaction files, the lower one-third of tank 241-C-104 contains CWP1 waste. Based on the HDW model prediction, the waste layers above the CWP1 waste include TH2 (Thoria 1970), CWP2, OWW3, PUREX zirconium cladding waste (CWZr1), SRR, and A1SltCk wastes. The CWP1 layer is approximately 76 to 90 kL (30 to 35 in.) thick. Only the lowest segment of cores 162 and 165 are expected to have intercepted the CWP1 layer. The high aluminum characteristics of CWP1 waste are present only in segment 4 of core 162 and in the lower half of segment 6 core 165 (Place 1998b). Only these segments will be included in

the CWP1 waste type template. The assumed date of analysis of this sample for decay purposes is September 1, 1996.

The HDW model waste transaction files indicate that tank 241-C-105 contains a 125 cm (49 in.) layer of CWP1 waste over a 14 cm (5 in.) layer of TBP. Only 60 percent of the CWP1 waste identified by the HDW model to be in tank 241-C-105 was verified by transfer records; the remaining 40 percent is an arbitrary assignment. The available sample data consists of the blended sample from two cores taken in 1986 (Weiss and Schull 1988a; Tusler et al. 1995). The solids and liquid sample data were combined by the ratio of 0.179-mL liquid per gram solids (the ratio of the lower two segments) and normalized by an assumed mass ratio of 1.2 (based on an assumed liquid specific gravity (SpG) of 1.10. The assumed date of analysis of this sample for decay purposes is May 1, 1986.

The HDW model waste transaction files indicate that tank 241-C-111 contains about 15 cm (6 in.) of CWP1 waste on top of approximately 41 cm (16 in.) of 1C1 and TFeCN wastes. The tank may also contain hot semiworks waste (HS). The high Pb, and Sr values found in the sample are characteristic of HS waste. The high nickel value is probably due to nickel ferrocyanide and is not typical of CWP1 waste (Simpson et al. 1995). The tank has a dished bottom and only the upper 8 inches of waste would have been accessible by the core sampler from the peripheral risers. The sample appears to be representative of CWP1 waste that has been contaminated with AR and TFeCN waste. The assumed date of analysis of this sample for decay purposes is May 1, 1994.

The composition for the CWP1 waste type template is based on sample values from tanks 241-C-103, 241-C-104, 241-C-105, and 241-C-111. The spreadsheet for the CWP1 waste type template composition (CWP1.xls) is stored on fileserver AP001\STDINV\RPP-8847\CWP1.xls.

A.4.3 CWP2 TANK WASTE LAYER ASSESSMENT

The tanks previously used as the basis for the CWP2 waste type template consisted of tanks 241-B-107, 241- B-109, 241-C-105, 241-S-111, and 241-U-109. Review of the sample data and process history indicates that tank 241-B-107 contains only a thin layer of CWP2 waste; the cladding waste in tank 241-B-108 is mixed with saltcake; the cladding waste in tank 241-C-105 is CWP1; and the cladding waste in tanks 241-S-111 and 241-U-109 is CWR1 waste.

Review of the Agnew et al. (1997a) indicates that of the sampled tanks, 241-B-107, 241-C-102, 241-C-107 and 241-T-102 were the most likely to have unadulterated layers of CWP2 waste.

Process history indicates that tank 241-B-107 received cladding waste supernatant from other 241-C Farm tanks in 1963 and contains CWP2 waste. Only one segment from the two core samples (218-1) is believed to be CWP2 waste because of the high Al content (Conner et al. 1998). The best estimate of the date of analysis for decay calculations is October 1, 1998.

Process knowledge indicates that tank 241-C-102 contains mostly CWP2 waste with some CWP1 waste, CWZr1 waste, and TH1 (Thoria 1966) sludge. Most of the analytical data for tank 241-C-102 came from a 1986 sample (Weiss and Schull 1988b). The balance of the data (Cl, SO_4, CO_3, and TOC) was obtained from an analysis of the archived sample (Hara 1990). The assumed date of analysis of this sample for radionuclide decay purposes is August 1, 1986.

Based on HDW model predictions, the core sample from tank 241-C-107 was expected to consist of about half CWP2 waste and half 1C, TBP, and HS/SRR wastes. However, core 71 apparently intercepted only the CWP2 layer in tank 241-C-107 (Place 1998c). Analytical data are based on the core 71 composite average for tank 241-C-107 upper sludge. The assumed date of analysis of this sample for decay purposes is April 1, 1995.

Although the HDW model predicts both MW and CWP2 waste in tank 241-T-102, the core sample intercepted only CWP2 waste (Baldwin et al. 1997b). The CWP2 layer may be adulterated with secondary Cs recovery waste. The best estimate of the date of analysis for decay calculations is June 1, 1993.

The composition for the CWP2 waste type template is based on sample values from tanks 241-B-107, 241-C-102, 241-C-107, and 241-T-102. The spreadsheet for the CWP2 waste type template composition (CWP2.xls) is stored on fileserver AP001\STDINV\RPP-8847\CWP2.xls.

A.5.0 WASTE CONCENTRATION SALTCAKE WASTES

There were four different waste evaporation methods: atmospheric-pressure 242-B and 242-T Evaporators; in-tank solidification tanks in the 241-BY Farm; self-boiling tanks in the 241-S, 241-SX, 241-A, and 241-AX Farms; and the 242-A and 242-S vacuum evaporators. Early evaporation processes were operated at atmospheric pressure and were atmospheric-pressure processes. The most recent evaporation processes operated under a vacuum, allowing evaporation to occur at lower temperatures. Saltcakes produced from the early evaporator processes can be frequently linked to specific supernatants. Later evaporation processes routinely combined supernatants from several sources before concentration.

A.5.1 BSltCk TANK WASTE LAYER ASSESSMENT

B saltcake (BSltCk) is derived from evaporator concentrates produced from the 242-B Evaporator from 1951 to 1954. Evaporator bottoms receivers include tanks 241-B-104, 241-B-105, 241-B-107, 241-B-108, and 241-B-109 (Agnew et al. 1996). These tanks collectively received about 27,400 kL (7,238 kgal) of evaporator concentrates, which produced 1,820 kL (480 kgal) of BSltCk solids (Agnew et al. 1997a). Tank 241-B-106 was used as the principal feed tank for the 242-B Evaporator and, as such, is expected to have a significant inventory of BSltCk. According to the HDW model, several other tanks also received BSltCk, including tanks 241-B-101, 241-B-102, 241-B-103, and 241-BX-112. The waste status and transaction records are inconsistent on this point because the 242-B Evaporator is reported to

have operated only from the fourth quarter of 1951 to the fourth quarter of 1954. Nevertheless, BSltCk concentrates are reported in the *Waste Status and Transaction Record Summary* (Agnew et al. 1997b) to have been sent to all of these alternate receivers (tanks 241-B-101, 241-B-102, 241-B-103, and 241-BX-112) during the fourth quarter of 1955 or more than a year after the 242-B Evaporator campaign was completed. These late transfers may have been created to balance the hypothetical waste inventories in the alternate receiver tanks (241-B-101, 241-B-102, 241-B-103, and 241-BX-112). According to the HDW model, tanks 241-B-101, 241-B-102, 241-B-103, and 241-B-105 contain about 1,116 kL (295 kgal) of BSltCk.

A review of the core sample profiles for tank 241-BX-112 (cores 118 and 119), together with HDW model, indicates that BSltCk was never added to this tank. The core sample profiles clearly show that a substantial amount of bismuth (15,000 to 20,000 μg/g) is present in all segments of both cores (Winkelman and Morris 1996). While bismuth is consistent with the expected composition of 1C2 waste, it is highly unlikely that this amount of bismuth would be found in any saltcake layer, including BSltCk. Core segment results for iron also appear to be more consistent with the composition profile that one might expect from sludge rather than saltcake. Iron concentrations vary from a low of 7,227 μg/g in segment 1A of core 118 to a high of 14,634 μg/g in segment 1 of core 119. These values are substantially higher than the values in other saltcakes (321 μg/g in A1SltCk, 1,330 μg/g in S1SltCk, 1,030 μg/g in S2SltCk, and 1,330 μg/g in T2SltCk, based on the analytical results from 21 tanks). In addition, the core sample profiles for Al, NO_2, NO_3, PO_4, SO_4, and ^{137}Cs do not indicate a break that one might expect during the transition from a sludge to a saltcake layer in the upper segments of both cores. Based on these results, it appears that the BSltCk layer was never added to this tank.

With the absence of adequate documentation relating to the transfer of 242-B Evaporator concentrates to tanks 241-B-101, 241-B-102, and 241-B-103, these tanks will not be included in the BSltCk waste type template.

Tank 241-B-104 received 1,100 kL (291 kgal) of BSltCk concentrate from the fourth quarter of 1952 to the third quarter of 1954. According to the HDW model, this tank contains 655 kL (173 kgal) of 2C1 waste, 41.6 kL (11 kgal) of 2C2 waste, 458 kL (121 kgal) of 1C2 waste, 15.1 kL (4 kgal) of unknown waste, and 230 kL (61 kgal) of BSltCk. Two core samples (cores 88 and 89) with 7 segments each were recovered from this tank in 1995 (Field et al. 1996a). Based on the estimated thickness of the BSltCk layer (56 cm [22 in]), this layer should be found in the uppermost segments of both cores. Segment 1 of core 88 contained 10 to 12 cm (4 to 5 in.) of solids, while segment 2 of that core contained 48 cm (19 in.) of solid material. About 35 cm (14 in.) of solids were recovered from segment 1 of core 89 and another 48 cm (19 in.) of material from segment 2 of that core. Based on these values, the BSltCk layer should be found in segments 1 and 2 of core 88 and in segment 1 of core 89. The core sample data, however, show that a substantial amount of bismuth is present in segments 1 and 2 of core 88 (13,780 to 16,100 μg/g) and in segment 1 of core 89 (8,620 to 10,800 μg/g). While high bismuth values are consistent with the expected composition of 1C2 waste, it is highly unlikely that this amount of bismuth would be found in any saltcake layer, including BSltCk. The concentration of iron is also unusually high in these segments (10,280 to 12,375 μg/g) and totally inconsistent with the amount of iron being found in other saltcakes. A1SltCk saltcake, for example, contains

only 321 μg/g of iron, while S1SltCk contains 1,330 μg/g, S2SltCk 1,030 μg/g, and T2SltCk 1,330 μg/g, based on the analytical results from 21 layers from 15 different tanks. The bismuth and iron concentrations are such that it is very unlikely that a separate saltcake layer exists in this tank. Because BSltCk concentrates were produced rather early in the evaporator campaigns, the evaporators were probably operated in a manner that would minimize or avoid any possible precipitation of solids in the evaporator or in the downstream receiver tanks. If this is true, evaporator concentrates may have been conveyed as supernatant to other tanks or eventually blended with other wastes in the tank and collected during one of the saltwell pumping campaigns.

Tank 241-B-106 was used as the principal feed tank for the 242-B Evaporator and, as such, is also listed as one of the principal receivers for BSltCk concentrates. According to the HDW model, this tank currently contains 439 kL (116 kgal) of BSltCk, which is consistent with the Hanlon (1999) inventory (442.8 kL [117 kgal]). Two push-mode core samples (cores 93 and 94) with two segments each were recovered from this tank in 1995 for safety screening purposes (McCain et al. 1996). Bismuth concentrations varied from 183 to 355 μg/g in segment 1 of both cores. These values are generally consistent with those from other saltcakes, ranging from 172 μg/g for A1SltCk saltcake, to 62 μg/g for S1SltCk, 66 μg/g for S2SltCk, and 60 μg/g for T2SltCk. High bismuth values (18,950 and 19,900 μg/g) in the bottom half of segment 2, however, indicate that other wastes, such as 2C1 waste from the 241-B-104/241-B-105/ 241-B-106 cascade, reside on the bottom this tank. While the analytical results for bismuth are consistent with those that might be expected for a saltcake layer in segment 1, the results for calcium, iron, and uranium are totally inconsistent with this interpretation. Calcium values in segment 1 of core 93 vary from 3,540 to 4,760 μg/g, compared to 227 to 434 μg/g in A1SltCk, S1SltCk, S2SltCk, and T2SltCk saltcakes. Similarly, iron values vary from 13,200 to 18,000 μg/g in both cores, compared to 321 to 1,330 μg/g in the previous saltcakes. Finally, the uranium values in segment 1 vary from 6,070 to 17,000 μg/g, compared to 371 to 915 μg/g from the reference saltcakes. Based on these comparisons, the waste material in segment 1 appears to be from a source other than BSltCk. Because of the uncertainty as to the source of this material, these segments will not be used to represent the composition of BSltCk.

Tank 241-B-107 received 3,395 kL (897 kgal) of evaporator bottoms from the 242-B Evaporator from the fourth quarter of 1952 to the fourth quarter of 1954. According to the HDW model, this tank contains 620.7 kL (164 kgal) of 1C1 waste but no BSltCk waste. Two core samples (cores 217 and 218) were obtained from this tank in 1997 (Conner et al. 1998). These cores were taken from opposite sides of the tank and show that 123 cm (48 in.) and 170 cm (67 in.) of waste is present below risers 6 (core 217) and 2 (core 218), respectively. Based on the average height of the sludge layers, the total inventory appears to be 602 kL (159 kgal), compared to an estimated inventory of 620.7 kL (164 kgal) from the HDW model and 624.5 kL (165 kgal) from Hanlon (1999). The core sample profiles provide a confusing picture as to the types of waste in this tank. Bismuth concentrations appear to be substantially higher in core 217 (4,500 to 20,900 μg/g) than in core 218 (122 to 11,200 μg/g). In this case, the bismuth rich layer seems to be intersected by segment 4 of core 218 and by the lower half of segment 1 and segments 2 and 3 of core 217. The bismuth rich layer is about 109 cm (43 in.) deep beneath riser 6 but only 45 cm (18 in.) deep below riser 2. Cerium can also be used along with bismuth as an indicator for

1C1 waste. The cerium profile generally matches the bismuth profile, which provides additional evidence that these layers consist of 1C1 waste and not BSltCk.

Segments 1 and 2 of core 218 and the upper half of segment 1 from core 217 from tank 241-B-107 represent a different waste. According to the relevant waste transaction records, only two different types of waste were transferred to this tank after the initial transfers of 1C1 waste. These types include BSltCk concentrates introduced to this tank from 1952 to 1955 and CWP2 waste transferred directly from PUREX, but also in the form of secondary transfers from tanks 241-C-101, 241-C-102, 241-C-103, and 241-C-106. The CWP2 waste would have produced high concentrations of aluminum in the CWP rich layers. This appears to be consistent with the high aluminum concentration found in the upper half of segment 1 of core 217 and in segments 1 and 2 of core 218. However, all but one of the aluminum values is also consistent with the aluminum concentrations found in A1SltCk, S1SltCk, S2SltCk, and T2SltCk saltcakes. Segments 1 and 2 of core 218 and the upper half of segment 1 of core 217 will, therefore, be used to represent BSltCk. The analytical results for Bi, Fe, and F, however, will not be included in the BSltCk waste type template because these values are not consistent with the results for A1SltCk, S1SltCk, S2SltCk, and T2SltCk. For example, these segments have an average Bi concentration of 2,713 μg/g, compared to an average Bi value of 89.3 μg/g for the other saltcakes. In this case, the BSltCk layer appears to have become contaminated with bismuth from the underlying 1C1 layer. These segments also have an average Fe concentration of 15,965 μg/g, compared to only 1,117 μg/g in the average of other saltcakes. The comparison for F is even more striking, with the BSltCk layer having 20,650 μg/g of F, compared to only 1,308 μg/g in the average of other saltcakes. Although it is possible that high Fe and F may have been derived from other wastes, such as CWP2 waste, the most likely source is cross-contamination from the adjacent 1C1 layer. Based on these considerations, the Bi, Fe, and F data from these segments will not be used for the BSltCk waste type template.

Tank 241-B-108 received 3,690 kL (975 kgal) of evaporator bottoms from the 242-B Evaporator in 1952-53. According to the HDW model, this tank currently contains 128.7 kL (34 kgal) of 1C1 waste and 227.1 kL (60 kgal) of BSltCk. Based on the projected volume of waste, the 1C1 layer should be 31.2 cm (12.3 in.) thick and the BSltCk layer 55.1 cm (21.7 in.) thick. Two separate cores were obtained from this tank in 1996 for safety screening purposes and for comparison of analyte concentrations to the HDW model (Schreiber et al. 1997). The first core (core 173) contained about 82.5 cm (32.5 in.) of waste from a sludge layer that was 142 cm (56 in.) deep. The second core (core 172) was somewhat less successful, recovering only 47 cm (18.5 in.) of waste from a sludge layer that was determined to be 91 cm (36 in.) deep. The 1C1 layer in this tank should be similar to the 1C1 layer in tank 241-B-107 (lower half of segment 1 and segments 2 and 3 of core 217 and segment 4 of core 218).

The segments from this tank (cores 172 and 173) have a much lower concentration of Bi and Fe than found in the 1C1 layer from tank 241-B-107. Based on this comparison, it appears that these cores did not intersect the 1C1 layer in tank 241-B-107 (which is consistent with the expected thickness of the 1C1 layer and the depth of the sludge layers that were evidently sampled). The BSltCk layer, therefore, will be represented by segments 1 and 2 of core 172 and by an average of the composite core results for core 173. Certain analytes found in these core

samples do not appear typical of BSltCk. The segments that are being used to represent the BSltCk layer have an average Bi concentration of 2,000 μg/g, which is much higher than the average of 89.3 μg/g for other saltcakes. These segments also have an average F concentration of 13,615 μg/g, compared to an average of only 1,308 μg/g for A1SltCk, S1SltCk, S2SltCk, and T2SltCk. Bismuth and fluoride results from these core samples will not be used in the BSltCk waste type template. The BSltCk layer appears to be contaminated with Bi and F from the underlying 1C1 layer. Several other analytes were also rejected for the BSltCk waste type template, including Ca, La, Pb, Sr, and Zr, because only high "less than" values are available for these analytes from the fusion analysis results. Nickel results were also found to be suspicious, and possibly too low, because these results are based only on water digestion samples. The total uranium values are based on kinetic phosphorescence measurements. The best estimate of the date of analysis for decay calculations is December 1, 1995.

Tank 241-B-109 received 5,007 kL (1,323 kgal) of BSltCk concentrate from the 242-B Evaporator from 1952 to 1954. According to the HDW model, tank 241-B-109 contains 317.9 kL (64 kgal) of BSltCk, 49.2 kL (13 kgal) of CWP2 waste, and 113.5 kL (30 kgal) of unknown waste. The BSltCk layer is approximately 77.3 cm (30.4 in.) thick, while the CWP2 and unknown layers are about 11.9 cm (4.7 in.) and 27.6 cm (10.9 in.) thick, respectively. Two cores (169 and 170), each consisting of two segments, were recovered from this tank in 1996 (Benar et al. 1997a). Core 170 was found to be nearly a full depth core of 119.2 cm (46.9 in.), while core 169 contained a much smaller amount of material. Because of the expected thickness of various layers, the BSltCk layer should be intercepted by the bottom segment (segment 3) of core 170, and possibly segment 2 of that core. Segment 2 of core 169 should also intercept the BSltCk layer. Analytical results from the core samples, however, indicate a different layer formation. The bottom segments of core 170 (segments 2C, 2D, and 3) contain a substantial amount of bismuth (7,660 to 13,100 μg/g), as well as significant amounts of iron (3,730 to 14,550 μg/g), nickel (4,150 to 5,750 μg/g), and fluoride (46,170 to 49,310 μg/g). These concentrations are significantly higher than the concentration of like components found in other saltcake templates (A1SltCk, S1SltCk, S2SltCk, T2SltCk, and T1SltCk). The bottom segments of core 170 also seem to have a much higher concentration of NO_3 and SO_4, but lower concentration of PO_4 than other segments of cores 169 and 170.

Based on this assessment, it appears that there are at least two distinct layers in tank 241-B-109. The upper layer is represented by segments 1, 2, and 2A of core 169 and segments 1 and 2 quarter A of core 170 and appears to be consistent with BSltCk or a blend of BSltCk and CWP2 waste. The lower layer represented by segment 2 quarters B, C, and D and the upper half of segment 3 from core 170 is probably related to 1C1 waste based on high concentrations of bismuth and fluoride. The upper layer, therefore, will be used to represent the BSltCk composition for tank 241-B-109. However, the bismuth and fluoride data from these segments will not be included in the BSltCk waste type template because this layer was probably contaminated with Bi and F from the underlying 1C1 layer. Iron, nickel, and silicon data from this layer will also be rejected because these contaminates were probably introduced from the CWP2 waste on top of the BSltCk layer, judging from the high Fe, Ni, and Si values associated with the CWP1 and CWP2 waste type templates. The potassium data is obviously erroneous (more than a million parts per million) and is due to KOH fusion of the sample. Total uranium

values also appear to be excessively high (28,375 μg/g), compared to a more typical value of 760 μg/g found in the average of four saltcakes (A1SltCk, S1SltCk, S2SltCk, and T2SltCk). For this reason, the segment uranium data will not be used in the BSltCk waste type template. Finally, the Pb data will be similarly rejected because it is based on high "less than" values from fusion analysis. The best estimate of the date of analysis for decay calculations is October 1, 1996.

The composition for the BSltCk waste type template is based on sample values from tanks 241-B-107, 241-B-108, and 241-B-109. The spreadsheet for the BSltCk waste type template composition (BSltCk.xls) is stored on fileserver AP001\STDINV\RPP-8847\BSltCk.xls.

A.5.2 BYSltCk TANK WASTE LAYER ASSESSMENT

The BYSltCk tank type template was originally developed from only three tanks in the 241-BY Farm (tanks 241-BY-105, 241-BY-106, and 241-BY-110). This waste type template, however, did not reference the identity of the segments that were used. In 1965, a prototype in-tank heater was initially installed in tank 241-BY-101 to concentrate the recirculating supernatant from other 241-BY Farm tanks. Later, in 1966, a second heater was placed in tank 241-BY-102, with tank 241-BY-103 being used as a primary feed tank. Finally, in 1966, a third heater was placed in tank 241-BY-112, with tank 241-BY-109 acting as the primary feed tank. During this third evaporation campaign, the heater in tank 241-BY-102 was used as a cooler with the hot concentrate from tank 241-BY-112 being routed to tank 241-BY-102. The ITS program ended in 1976 with 143,830 kL (37,983 kgal) of volume reduction and about 14,712 kL (3,887 kgal) of saltcake (BYSltCk) being formed.

According to the HDW model, significant inventories of BYSltCk currently reside in tanks 241-B-112, 241-BX-106, 241-BX-110, 241-BX-111, 241-BY-101, 241-BY-102, 241-BY-103, 241-BY-104, 241-BY-105, 241-BY-106, 241-BY-107, 241-BY-108, 241-BY-109, 241-BY-110, 241-BY-111, and 241-BY-112. About 93 percent of this inventory apparently is consolidated in the BY Farm tanks. Of the 16 tanks with BYSltCk, core samples have been taken from 14 tanks representing 98.9 percent of the total BYSltCk inventory. Auger samples from the other two tanks were taken for safety screening purposes and were analyzed for a limited number of analytes. Because of the extensive amount of sampling that has been done, the BYSltCk waste type template will be used by the BBI to fill in when analytical data for BY saltcake (BYSltCk) layers are missing.

For those tanks with in-tank heaters (241-BY-101, 241-BY-102, and 241-BY-112), in-tank heating and cooling processes have apparently produced higher concentrations of F, Fe, Cr, Ni, PO_4, Si, and SO_4 (and lower concentrations of NO_3) than found in other BY-Farm tanks. In tank 241-BY-101, for example, concentrations of Cr, F, Si, and especially PO_4 were found to be relatively high in tank 241-BY-102; F, Fe, Ni, PO_4, Si, and SO_4 are much higher than other BY-Farm tanks. And in tank 241-BY-112, several analytes, including F, Fe, Cr, PO_4, Si, and SO_4, were found to be high, demonstrating that in-tank heating and cooling processes have altered the composition profile in these tanks. Because of these differences, the BYSltCk waste type template will be developed from tanks that were not equipped with in-tank heaters.

According to the HDW model, tank 241-BY-104 contains about 567 kL (150 kgal) of PFeCN2 waste and 666 kL (178 kgal) of BYSltCk. Two core samples (cores 116 and 117) were recovered from this tank in late 1995 (Benar et al. 1996). Core 116 consists of 8 segments, with segment 1 being empty. The total length of this core was found to be 337.8 cm (133 in.), with most of the material that was recovered being described as wet or dry salt (except for segment 8 which was described as soft sludge). Core 117 is comprised of 5 segments containing approximately 192.6 cm (75.84 in.) of waste. Apparently, the surface of the waste has subsided in the tank center such that core 117 contained substantially less material than core 116. The composition profiles are such that two distinct layers of waste appear to be present in this tank. These layers are distinguished from one another by substantial differences in the concentration of Al, Bi, Ca, Fe, Pb, Mn, Ni, and U. The bottom layer represented by sub-segments 7C and 7D, and segment 8 of core 116 apparently consists of PFeCN2 waste, TBP waste, and possibly MW. The Bi concentration in the upper layer is about 40 μg/g, compared to 8,800 to 14,000 μg/g in the bottom layer. The iron concentration varies from about 500 μg/g in the upper layer to 32,400 μg/g in the bottom layer, while the uranium concentration varies from about 500 μg/g in the upper layer to 23,900 μg/g in segment 8 of core 116. Interestingly enough, high iron and uranium concentrations were found only at the bottom of core 116 and not at the bottom of core 117, indicating perhaps a small pocket of MW. The upper layer represented by segments 2, 3, 4, 5, and 6 of core 116 and by segments 1, 2, and 3 of core 117 apparently consists of BYSltCk. This layer will be used to construct the BYSltCk waste type template. The best estimate of the date of analysis for decay calculations is December 1, 1995. Segment 7 of core 116 and segments 4 and 5 of core 117 were not used because these segments may overlap both layers (although the results for segments 4 and 5 were similar to other BYSltCk segments but were found to be suspect because of high NO_3).

Tank 241-BY-105 is reported by the HDW model to contains 60.5 kL (16 kgal) of MW, 537.5 kL (142 kgal) of PFeCN2, 1,275.5 kL (337 kgal) of BYSltCk, and 30.3 kL (8 kgal) of CEM (Portland cement). Only one partial core sample was obtained from this tank in 1995 (Simpson et al. 1996b). Because of core sampling difficulties related to the cement layer in the tank, coupled with safety related issues with rotary drilling, sampling efforts were terminated with a partial core (segments 1, 1A, EAR, 1B, 1R, 2, EAR, 2R, and 3). Tank farm process history records show that numerous transfers occurred between tank 241-BY-105 and other tanks in the BY-Farm as part of the ITS program from 1966 to 1976. During the first quarter of 1977, Portland cement was added to this tank, followed in 1982 by saltwell pumping of interstitial liquid and supernatant to tank 241-AW-102. Based on this transaction history, the upper segments of core 108 should represent the composition of BYSltCk, except for Al, Ca, Fe, Si, and SO_4 that may be present from the Portland cement. While the physical description of the core segments suggests that these samples consist of Portland cement, the composition profiles for Al, Ca, Fe, Si, and SO_4 suggest otherwise. In all cases, the concentrations of Al, Ca, Fe, Si, and SO_4 were found to be lower than similar components in the BYSltCk layer in tank 241-BY-104. Based on this comparison, the core sample segments from tank 241-BY-105 appear to represent BYSltCk, and for this reason, will be included in the BYSltCk waste type template. The best estimate of the date of analysis for decay calculations is November 1, 1995.

Tank 241-BY-106 is assumed to have 204 kL (54 kgal) of PFeCN1 waste, 167 kL (44 kgal) of PFeCN2 waste, and 2,059 kL (544 kgal) of BYSltCk. Three core samples (cores 64, 65, and 121) have been recovered from this tank. Core 65 is of no value because all of the segments except one (segment 13) contained wash water or were found to be empty. Based on the amount of material predicted for each waste layer, the PFeCN1 and PFeCN2 layers should be represented by the bottom two segments of cores 64 and 121 (and by segment 13 of core 65). Analytical results are currently available only for core 121 (Bell et al. 1996). Because the bottom 144.5 cm (56.9 in.) of waste were not sampled by this core (segments expected to contain PFeCN1 and PFeCN2 wastes), all of the segments that were recovered are assumed to represent BYSltCk. This assumption is supported by the composition profiles for this waste which show that there is very little variation across the segments in the concentration of Al, Bi, Ca, Fe, Mn, Na, Ni, Si, Sr, U, Cl, F, NO_3, NO_2, PO_4, and SO_4. Based on these findings, the composite average of core 121 will be used to represent the analytes in the BYSltCk waste type template. The best estimate of the date of analysis for decay calculations is February 1, 1996.

According to the HDW model, tank 241-BY-107 currently contains 90.8 kL (24 kgal) of 1C2 waste, 75.7 kL (20 kgal) of TBP waste, 162.7 kL (43 kgal) of PFeCN1, 113.5 kL (30 kgal) of PFeCN2, and 563.9 kL (149 kgal) of BYSltCk. Based on these estimates, the 1C2 layer should be 22 cm (8.7 in.) thick, the TBP layer 18.3 cm (7.2 in.) thick, the PFeCN1 and PFeCN2 layers 39.6 cm (15.6 in.) and 27.7 cm (10.9 in.) thick, respectively, and the BYSltCk layer 137.2 cm (54 in.) thick. The BYSltCk layer, therefore, should be found in the top three segments of the cores. Two core samples (cores 151 and 161) were recovered from this tank in 1996 (McCain et al. 1997). Because of high sampling pressures encountered, only four segments were recovered from each core, although six segments should have been recovered if the core samples had reached the bottom of the tank. The segment composition sequences are such that two distinct layers seem to be present in the material that was recovered. Based on the segment concentrations for Fe, Mn, Ni, Si, and U, the material in segments 1 through 3 and in the upper half of segment 4 consist of BYSltCk, while the lower half of segment 4 consists of sludge which presumably matches the composition of PFeCN waste. The upper three segments of both cores will be used to construct the BYSltCk waste type template. Material from the upper half of segment 4 will not be used because some of this waste may have commingled with the underlying sludge layer. The best estimate of the date of analysis for decay calculations is September 1, 1996.

Tank 241-BY-108 is assumed to contain 503.4 kL (133 kgal) of PFeCN1 waste, 121.1 kL (32 kgal) of PFeCN2, and 238.4 kL (63 kgal) of BYSltCk. These layers correspond to about 122.4 cm (48.2 in.) of PFeCN1 waste, 29.4 cm (11.6 in.) of PFeCN2 waste, and 57.9 cm (22.8 in.) of BYSltCk. Based on these estimates derived from the HDW model, the BYSltCk layer should be found in the upper two segments of the cores that were recovered from this tank. Altogether, three core samples were obtained from this tank in 1995 (cores 98, 99, and 104). The segment concentration sequence for Fe, Pb, Ni, Sr, U, NO_3, PO_4, ^{137}Cs, $^{239/240}Pu$, and ^{90}Sr shows that there are two distinct layers in this tank (Baldwin et al. 1996a). The upper layer represented by segments 1 through 3 of cores 98 and 99 and segments 1 through 4 of core 104 is consistent with the expected composition of BYSltCk which has lower concentrations of Fe, Pb, Ni, Sr, U, PO_4, ^{137}Cs, $^{239/240}Pu$, and ^{90}Sr, but higher concentrations of NO_3. The lower layer that appears in

segment 4 of cores 98 and 99, as well as segment 5 of core 104, has significantly higher concentrations of Fe, Pb, Ni, Sr, U, PO_4, ^{137}Cs, $^{239/240}Pu$, and ^{90}Sr, all of which would be expected in the PFeCN layer. Based on these considerations, segments 1 through 3 of cores 98 and 99, together with segments 1 through 3 of core 104 will be used to construct the BYSltCk waste type template. Segment 4 of core 104 will not be included in the BYSltCk layer because of possible commingling with segment 5 of that core. The best estimate of the date of analysis for decay calculations is September 1, 1995.

Tank 241-BY-109 is assumed to have 136.2 kL (36 kgal) of unknown waste tentatively identified by the HDW model as MW and 1,464.8 kL (387 kgal) of BYSltCk. Two core samples were recovered from this tank in 1997 (cores 201 and 203). Core 201 consisted of 6 segments, while core 203 was comprised of 7 segments. The core sample composition sequence shows that two distinct layers exist in this tank (Jo et al. 1998b). The bottom layer is defined by the lower half of segment 7 and is distinguished from the upper layers by higher concentrations of Al and Mn, but significantly lower concentrations of Na, Sr, U, Zr, F, and SO_4. This layer in all likelihood does not consist of MW, as alleged in the HDW model, because the uranium concentration in this layer is much lower than would be expected in metal waste. The upper segments are generally uniform in composition and are assumed to be BYSltCk. The composition of this upper layer will be based on the average solid segment data for cores 201 and 203. Only a small amount of error is likely to be introduced by this approach because the average solid segment data were generated from 19 whole or partial segments from both cores, only one of which is from the unknown sludge layer (lower half of segment 7 from core 203). Based on the extensive amount of data from the BYSltCk region, the average solid segment data are deemed to be acceptable for the BYSltCk waste type template. The best estimate of the date of analysis for decay calculations is August 1, 1997.

Tank 241-BY-110 presumably contains 140 kL (37 kgal) of 1C2 waste, 420 kL (111 kgal) of PFeCN1 waste, 159 kL (42 kgal) of PFeCN2 waste, and 779.7 kL (206 kgal) of BYSltCk. Based on these estimates which were derived from the HDW model, the BYSltCk layer should be 189.5 cm (74.6 in.) deep, the PFeCN2 layer 38.6 cm (15.2 in.) deep, the PFeCN1 layer 102 cm (40.2 in.) thick, and the 1C2 layer about 34 cm (13.4 in.) thick. A total of nine core samples were recovered from this tank in 1995 (cores 92, 95, 96, 101, 103, 106, 107, 109, and 113). Cores 92, 95, 96, and 109 were obtained from riser 12B, while cores 101, 103, 106, and 107 were taken from riser 7. Core 113 is the only core obtained from riser 4. The average composition for the saltcake layer is provided in Table 4-3 of Schreiber et al. (1996). This information was developed from all of the segments of cores 92, 95, 96, 106, and 109. Saltcake estimates were also derived from segments 1 through 8B of core 101, segments 1 through 8C of core 103, segments 1 through 7D of core 107, and segments 1 through 6 of core 113 (as described in Table 4-2 of the Schreiber report). These saltcake values will be used to develop the BYSltCk waste type template. The best estimate of the date of analysis for decay calculations is December 1, 1995.

Tank 241-BY-111 is thought to contain 98.4 kL (26 kgal) of MW and 1,638.9 kL (433 kgal) of BYSltCk. These inventory estimates, developed from the HDW model, correspond to 23.9 cm (9.4 in.) of MW and 398.4 cm (156.8 in.) of BYSltCk. Two core samples (cores 168 and 171) were taken from this tank in 1996 (Jo et al. 1997). Because of sampling problems, core 168 recovered only 6 segments, while the second core, core 171, recovered 7 segments. It was planned that both cores would consist of 9 segments, representing the full depth of waste in the tank. According to the core sample profiles, core 168 recovered the upper 241-cm (94.84-in.) of material, while core 171 sampled the upper 337 cm (132.76 in.) of waste. Both cores should, therefore, have only intercepted the BYSltCk layer in this tank.

The core segment compositions generally support this assessment except for segments 1 through 4 of core 171. These segments are unusual because they contain substantially higher concentrations of Al, Fe, Si, and Sr, but lower concentrations of Na, NO_2, NO_3, PO_4, and SO_4 than the adjacent segments of cores 168 and 171. These concentrations may be due to PUREX coating wastes added directly to this tank in 1967 or indirectly through tank 241-C-102 in 1966. The ITS program implemented in 1966 is not especially well documented in that large transfers of BY saltcake concentrates (644 kgal of such concentrates) are alleged to have occurred at the end of the ITS program in 1976. It is possible that this late transfer could have added contaminated wastes from other 241-BY Farm tanks to tank 241-BY-111, therefore, biasing the normal concentration profile of other BYSltCk wastes. The Fe concentrations (13,520 to 36,700 μg/g) and silicon concentrations (274,000 to 85,400 μg/g) associated with segments 1 through 4 are clearly large and totally inconsistent with the concentration of similar analytes in other BY saltcake tanks. Based on these discrepancies, segments 1 through 4 of core 171 will not be included in the BYSltCk waste type template. Other segments, including segments 1 through 6A of core 168 and segments 5 through 7 of core 171, will be averaged to produce a representative composition for the BYSltCk layer in this tank. The best estimate of the date of analysis for decay calculations is November 1, 1996.

The composition of the BYSltCk waste type template is based on sample values from tanks 241-BY-104, 241-BY-105, 241-BY-106, 241-BY-107, 241-BY-108, 241-BY-109, 241-BY-110, and 241-BY-111. The spreadsheet for the BYSltCk waste type template composition (BYSltCk.xls) is stored on fileserver AP001\STDINV\RPP-8847\BYSltCk.xls.

A.5.3 T1SltCk TANK WASTE LAYER ASSESSMENT

The T1SltCk group consists of those tanks that received evaporator bottoms from the first 242-T Evaporator campaign (1951 to 1955). The primary receiver tanks include 241-T-109, 241-TX-103, 241-TX-113, 241-TX-116, 241-TX-117, and 241-TY-102. These tanks collectively received some 35,130 kL (9,280 kgal) of evaporator concentrate, with tanks 241-TX-116 and 241-TX-117 receiving more than 80 percent of the total inventory (10,167 kL [2,686 kgal] in tank 241-TX-116 and 18,438 kL [4,871 kgal] in tank 241-TX-117) (Agnew et al. 1996). Several samples have been taken from this group, including one core from tank 241-TX-116 in 1977, another core from tank 241-TY-102 in 1986, and several auger samples from tanks 241-T-108 and 241-T-109 in 1995. The core from tank 241-TY-102 and the auger

samples from tank 241-T-108 may be of only limited value because these samples represent a blend or composite of several wastes (T1SltCk and T2SltCk in tank 241-TY-102 and T1SltCk and 1C1 in tank 241-T-108). Auger samples present an additional problem in that sample recoveries are such that the various layers may not be accurately represented in the final blend.

According to the HDW model, tank 241-T-108 is assumed to have 79.5 kL (21 kgal) of 1C1 waste and 87.1 kL (23 kgal) of T1SltCk, while tank 241-T-109 contains only one waste (87.1 kL [23 kgal] of T1SltCk). Based on this assessment, it appears that the auger samples from tank 241-T-109 could be used for the T1SltCk tank waste layer composition. However, it appears that the samples from tanks 241-T-108 and 241-T-109 (Baldwin et al. 1996b and Brown et al. 1996) are not consistent with the analytical results from tank 241-TX-116 (Horton 1977). Tank 241-T-108 and 241-T-109 wastes contain significantly higher concentrations of PO_4 and F than the waste in tank 241-TX-116 (125,000 and 246,000 μg/g of PO_4 compared to 12,167 μg/g in tank 241-TX-116, and 10,700 and 13,000 μg/g F compared to 3,087 μg/g in tank 241-TX-116). Tank 241-T-109 also contains a much lower concentration of NO_3 than found in tank 241-TX-116, which is consistent with a large inventory of sodium phosphate in this tank. As a final note, it also appears that tank 241-TX-116 contains much more SO_4 than tanks 241-T-108 and 241-T-109 (34,500 μg/g SO_4 compared to only 1,110 and 516 μg/g in tanks 241-T-108 and 241-T-109, respectively).

These discrepancies are important because they indicate that other wastes, such as 1C1 and TBP supernatants, have had a significant effect on the composition of the solids in both tanks. 1C1 supernatant was routed to the 241-T-108/241-T-109 cascade in 1945-1946 and TBP supernatant to this cascade in 1952-1953. These transfers are evidently linked to the precipitation of sodium phosphate or sodium-fluoro-phosphate double salts in these tanks. The main T1SltCk receivers (tanks 241-TX-116 and 241-TX-117) did not receive significant transfers from the $BiPO_4$ or TBP processes (1C, 2C, 224, or TBP waste). Although tank 241-TX-117 did receive some 2C waste in 1951, these transfers only occurred after the waste had been stored in the primary receiver tanks (241-T-105 and 241-T-106) for several years (well after the initial solids would have precipitated from this waste). Because of these differences, the auger samples from tanks 241-T-108 and 241-T-109 may not be especially representative of the T1SltCk wastes in tanks 241-TX-116 and 241-TX-117 and, therefore, will not be included in the T1SltCk waste type template.

The T1SltCk waste type template will mostly be based on the composition of tank 241-TX-116 waste. A breakdown of the 1977 core shows that T1SltCk is represented by segments 6, 7, 9, and 10. Segments 1 through 4 of this core represent the T2SltCk layer. Most analytes in the T1SltCk waste layer will be derived from segments 6 through 10 of this core. The T2SltCk waste type template, however, will be used to supply missing analytes such as Bi, Ca, Cl, K, La, Mn, Ni, Pb, Sr, TOC, and Zr to the T1SltCk layer of this tank. The Fe and Si values from segments 6 through 10 are suspiciously high, and appear to be values that might be expected from a sludge rather than saltcake (15,942 μg/g and 7,250 μg/g, respectively). These values will be replaced with comparable values from the T2SltCk waste type template. The uranium concentration in segments 6 through 10 appears to be unusually low for saltcake and will be replaced with the T2SltCk uranium value for these segments. The T2SltCk waste type template values for total

alpha and ^{241}Am will also be used to supply radionuclides missing from the analysis of the T1SltCk layer for this tank. Physical parameters such as percent water and bulk density will also be based on the T2SltCk values developed from tanks 241-TX-104, 241-TX-116, 241-U-103, and 241-U-105. The best estimates of the dates of analysis for decay calculations are August 1, 1995 for 241-T-108, September 1, 1995 for 241-T-109, and March 1, 1977 for TX-116.

The composition of the T1SltCk waste type template is based on sample values from tanks 241-T-108, 241-T-109, and 241-TX-116. The spreadsheet for the T1SltCk waste type template composition (T1SltCk.xls) is stored on fileserver AP001\STDINV\RPP-8847\T1SltCk.xls.

A.5.4 T2SltCk TANK WASTE LAYER ASSESSMENT

T2SltCk is occasionally referred to as SMMT2 in BBI reports. The HDW model uses the SMMT2 nomenclature. T2SltCk waste is comprised of the solids produced from the 242-T Evaporator concentrates during the period from 1965 to 1976. The 242-T Evaporator bottom receivers included one tank from the 241-S Farm, two tanks from 241-SX Farm, four tanks from 241-T Farm, 17 tanks from 241-TX Farm, two 241-TY Farm tanks, and five tanks from 241-U Farm. These tanks collectively contain about 19,600 kL (5,180 kgal) of T2SltCk solids, with about 77 percent of this inventory (15,100 kL [3,997 kgal]) residing in tanks that have not been sampled in the 241-T and 241-U Farms. The T2SltCk waste composition is important because it represents a significant fraction of the unsampled saltcake inventory in the 241-TX and 241-U tank farms. The previous T2SltCk waste type template was developed from core sample data from tanks 241-TX-116, 241-U-102, and 241-U-105. This waste type template, however, did not reference the identity of the segments that were used to construct the composition. Since that time, new analytical results have been obtained from other T2SltCk receivers, tanks 241-TX-104 and 241-TX-113. These cores (cores 230 and 231 from tank 241-TX-104 and cores 253 and 258 from tank 241-TX-113) should also be included in the T2SltCk waste type template.

In tank 241-TX-104, segments 1, 2, and 2A from cores 230 and 231 represent the composition of the T2SltCk waste in this tank (Field et al. 1999). According to the HDW model, the current inventory consists of 68 kL (18 kgal) of metal waste (MW) and 174 kL (46 kL) of T2SltCk saltcake. Because the metal waste was sluiced out in 1954-55 and once again in 1956, most, if not all, of remaining waste probably consists of T2SltCk saltcake (Rhodenhizer 1987; Agnew et al. 1997a). This assignment is consistent with the known waste transaction records for this tank (*Waste Status and Transaction Record Summary*, Agnew et al. 1997b). Other records also show that 6,612 kL (1,747 kL) of T2SltCk concentrates were added to tank 241-TX-104 during the first quarter of 1976 (Agnew et al. 1996). The analytical results for cores 230 and 231 are currently stored in the TCD. This data can be sorted by waste type, analysis method, and by analyte to develop the composition profiles needed for the T2SltCk waste layer composition. The composition profiles for Cl, Cr, Fe, Ni, Na, NO_3, NO_2, PO_4, and SO_4 are such that there is no clear evidence of more than one waste type in the tank. The profile for uranium, however, does suggest some enrichment at the bottom of the tank that would be consistent with a small residual amount of metal waste in the tank. The top layer appears to be enriched in sodium aluminate and

sodium oxalate, both of which could have precipitated from the evaporator concentrates added to this tank during the first quarter of 1976. For all practical purposes, the composite sample appears to be acceptable for the T2SltCk waste type template. Because of the preponderance of T2SltCk waste expected in this tank, the segment compositions for cores 230 and 231 were averaged, on a weighted basis, to produce a composite composition for the T2SltCk waste layer. The tank 241-TX-104 waste was found to have an average density of 1.62 g/mL, with 45.54 percent moisture content. The best estimate of the date of analysis for decay calculations is March 1, 1998.

According to the HDW model, tank 241-TX-113 received 14,500 kL (3,845 kgal) of T2SltCk concentrates from the 242-T Evaporator from 1966 through 1971. This tank currently contains about 2,300 kL (607 kgal) of waste based on manual tape measurements taken on April 16, 1996. The HDW model predicts that 692 kL (183 kgal) of this waste is 1C2 sludge and the remaining 1,604 kL (424 kgal) is T2SltCk. Two core samples were obtained from this tank in 1998-1999 (cores 253 and 258). Core 253 recovered small amounts of waste in the first 10 segments but the bottom full segment and another partial segment were not recovered because of sampling problems in the field. Small amounts of material were also recovered from the top seven segments of core 258 before the sampling effort was terminated. Together, these segments represent about 530 cm (209 in.) of waste, with approximately 56 cm (22 in.) of unsampled material at the bottom of the tank. The segment level concentrations for Cl, Cr, Fe, NO_2, and PO_4 are such that there is no clear evidence of layers that would be consistent with more than one waste type in the tank. With 1C2 sludge, one would expect to see much higher concentrations of Bi (18,700 μg/g), Fe (10,900 μg/g), and PO_4 (71,800 μg/g) than is found in T2SltCk (60 μg/g Bi, 1,330 μg/g Fe, and 15,700 μg/g PO_4). Because there is no evidence of the Bi, Fe, and PO_4 rich layer expected from 1C2 sludge, it appears that all of the recovered waste is probably T2SltCk. The core sample analysis, however, also reveals significant differences between the two cores. Core 253 was taken from riser 3 located halfway toward the center of the tank, while core 258 was obtained from riser 5 which is closer to the center of the tank. The Al, NO_3, and CO_3 segment compositions differ by nearly a factor of three between the two cores. More importantly, the SO_4 concentration in core 253 is approximately fifty times higher than core 258 (159,000 versus 2,870 μg/g). These differences appear to be from concentration in the 242-T Evaporator during the period from 1966 through 1971. The high SO_4 concentration found in core 253 is consistent with formation of a sodium nitrate-sulfate double salt, which may have been caused by multiple evaporation cycles. The segment compositions for cores 253 and 258 were averaged to produce a composition for the T2SltCk waste layer. The tank 241-TX-113 waste was found to have 8.92 wt% moisture content. The best estimate of the date of analysis for decay calculations is April 1, 1999.

Tank 241-TX-116 received 9,900 kL (2,615 kgal) of T2SltCk concentrates from 1965 to 1969 and 10,160 kL (2,684 kgal) of T1SltCk concentrates from 1951 through 1953 (Agnew et al. 1996). These layers are observable in the 1976 core sample from this tank (Horton 1977). This core consists of 294 cm (116 in.) of T1 saltcake (T1SltCk) found in segments 1 through 4, and 284 cm (112 in.) of T2SltCk saltcake in segments 6 through 10. The T2SltCk and T1SltCk layers can be distinguished from one another by sharp differences in the Al, Na, NO_3, NO_2, SO_4, and ^{90}Sr concentrations. Segments 6 through 10 were averaged to produce a composite analysis

for the T2SltCk waste layer. The analytical results for Fe, Si, and ^{99}Tc were not included in this waste layer composition for several reasons. First, the analytical results for Fe, Si, and ^{99}Tc are suspect because these values are totally inconsistent with comparable values from T2SltCk waste layers found in other tanks (241-TX-104, 241-U-102, and 241-U-105). The Fe value is in the range that is expected for a sludge rather than saltcake sample. The Si value is also high relative to other saltcake samples, and this may be due to the Parr bomb procedure used in the late 1980's (high Si results were also obtained for tanks 241-C-106 and 241-C-104 by Parr bomb procedure during this period). Secondly, the analytical results for Fe, Si, and ^{99}Tc were not used because these results were obtained from a sample that was not analyzed under current quality control procedures (with duplicates, standards, and spike recovery samples). The remaining analytes were used to construct the T2SltCk waste layer composition for this tank. Waste in tank 241-TX-116 has a density of 1.7 g/L and measured moisture content of 28.4 wt%. The best estimate of the date of analysis for decay calculations is March 1, 1977.

Several cores have also been obtained from tank 241-U-102. This tank received 677 kL (179 kgal) of T2SltCk concentrates during the first quarter of 1976 and currently has a projected inventory 162 kL (43 kgal) of metal waste (or REDOX waste), 810 kL (214 kgal) of T2SltCk solids, and 374 kL (99 kgal) of S2SltCk solids (Agnew et al. 1996 and 1997b). The two 1996 cores from this tank (cores 143 and 144) contain a mixture of soft sludge and moist saltcake, with two distinct layers (Hu et al. 1997). Because the cores did not intercept the sludge layer expected at the bottom of the tank, these cores are believed to represent only the two uppermost saltcake layers in the tank. The S2SltCk layer can be distinguished from the lower T2SltCk layer by the PO_4 and SO_4 concentrations. On the average, the T2SltCk layer contains about 5,030 μg/g PO_4 and 15,560 μg/g SO_4, while the S2SltCk layer has 24,850 μg/g PO_4 and 12,370 μg/g SO_4. Other analytes, including Na, NO_2, and NO_3, have concentrations that are generally similar for both layers. Segments 1 and 2 are believed to represent the S2SltCk layer and segments 3, 4, 5, and 6, together with sub-segments 5A, 5B, and 6A, the T2SltCk layer. The analytical results were averaged, with equal weight being given to each segment and sub-segment, to estimate the composition for the T2SltCk layer. Because the segment and sub-segment cores commingled to a certain extent, this duplication was resolved by simply averaging the results for each segment and sub-segment. The 241-U-102 samples had an average density of 1.665 g/mL and average moisture content of 31.67-wt% moisture. The best estimate of the date of analysis for decay calculations is June 1, 1996.

Tank 241-U-105 received 1,050 kL (277 kgal) of T2SltCk concentrates in early 1976, 2,036 kL (538 kgal) of S1SltCk concentrates in 1977, and another 1,786 kL (472 kgal) of S2SltCk concentrates in 1978 (Agnew et al. 1996). According to the HDW model, this tank contains 121 kL (32 kgal) of metal waste, 280 kL (74 kgal) of T2SltCk solids, and 1,040 kL (275 kgal) of S2SltCk solids (actually a composite of S1SltCk and S2SltCk solids following the established convention in the HDW model [Agnew et al. 1997a]). Three complete core samples (cores 131, 133, and 138) were obtained from this tank (Brown and Franklin 1996). These cores indicate a waste level that is consistent with waste height measurements for this tank (411 cm [162 in.]). The first segment contained liquid, while segments 2 through 7 consist of a mixture of soft saltcake and sludge (S2SltCk waste), and segments 8 and the upper half of segment 9 of soft saltcake and sludge (T2SltCk layer). The lower half of segment 9 is described as soft sludge

(potentially metal waste). The segment compositions, however, support a slightly different waste layering. The bottom half of segment 9 clearly consists of sludge based on the concentrations of Al, Fe, Mg, Mn, PO_4, and especially U. The Al, PO_4 and U concentrations in the lower half of segment 9 support conclusion that this layer consists of metal waste and secondary coating waste from tank 241-U-108. Segment concentrations of Al, Cl, PO_4, SO_4, ^{154}Eu, and total alpha in the remaining segments (segments 1 through the upper half of segment 9) do not allow identification of a distinct boundary between the S2SltCk and T2SltCk layers. These two layers appear to be essentially identical to one another. Based on the chronological ordering of the saltcake layers, the averaged analyses of the upper and lower portions of segment 8 were chosen to represent the composition of the T2SltCk layer. This segment has an average density of 1.73 g/mL and contains 24.03 wt% moisture. While most upper segments in this tank are described as consisting of a mixture of saltcake and sludge, the waste transaction records clearly support the precipitation of saltcake layers from the 242-T and 242-S Evaporator concentrates. The best estimate of the date of analysis for decay calculations is April 1, 1996.

Tanks 241-TY-102 and 241-TY-103 were initially considered as candidates for the T2SltCk waste type template, but the composite core samples from these tanks were rejected because of other wastes in these tanks. Tanks 241-S-107, 241-SX-103, and 241-SX-105 also received T2SltCk concentrates, but the structures of the sludge layers in tank 241-S-107 were found to be too complicated for identification of the T2SltCk layer (only one of eight layers in the tank representing 10 percent of the waste). Core sample data for tanks 241-SX-103 and 241-SX-105 were not published at the time of this analysis and, therefore, could not be used. Tanks 241-SX-103 and 241-SX-105 also contain large amounts of RSltCk and S2SltCk, which would likely make identification of the T2SltCk layer difficult, if not impossible. Tanks 241-T-105, 241-T-107, 241-T-108, and 241-T-112 are other possible candidates. However, very little T2SltCk is believed to exist in these tanks, based on the other identified waste types (1C1, 1C2, 2C1, 2C2, and T1SltCk) and projected waste inventories in these tanks. Finally, tanks 241-U-101, 241-U-103, and 241-U-104 were also listed as T2SltCk concentrate receivers. The amount of T2SltCk concentrate added to tanks 241-U-101 and 241-U-104 appears to be very small and of little use in establishing a composition for the T2SltCk layer. The amount added to tank 241-U-103 is appreciable (1,237 kL [327 kgal]), but the identified saltcake layers in the HDW model principally consist of S1SltCk and S2SltCk saltcakes, and not the T2SltCk layer that is being sought. Aside from these tanks, most of the T2SltCk inventory appears to be in the TX-Farm tanks that have not been sampled and, therefore, cannot be used for development of the T2SltCk tank waste layer composition.

The composition of the T2SltCk waste type template is based on sample values from tanks 241-TX-104, 241-TX-116, 241-U-102, and 241-U-105. The spreadsheet for the T2SltCk waste type template composition (T2SltCk.xls) is stored on fileserver AP001\STDINV\RPP-8847\T2SltCk.xls.

A.5.5 A1SltCk TANK WASTE LAYER ASSESSMENT

A1SltCk waste is comprised of the solids produced from 242-A Evaporator during the period from 1976 to 1980. This waste was routed to 26 different receivers in the 241-A, 241-AW, 241-AX, 241-AY, 241-AZ, 241-BX, 241-BY, and 241-C Farms (Agnew et al. 1996). These tanks collectively received about 104,000 kL (27,500 kgal) of A1SltCk concentrate and contain 7,150 kL (1,890 kgal) of A1SltCk solids, with about 38 percent of this inventory (2,736 kL [723 kgal]) residing in tank 241-AX-101. Tank 241-AX-101, however, has not been sampled. Waste samples from six of the tanks that contain this waste were examined for A1SltCk waste layers. These tanks include 241-A-101, 241-A-102, 241-A-103, 241-A-106, 241-AX-102, and 241-AX-103. The auger sample from tank 241-AX-102 was excluded from consideration because only a limited number of analytes were reported and the TOC and total alpha concentrations were significantly higher than TOC and total alpha concentrations found in other A1SltCk layers. According to the HDW model, very little A1SltCk waste currently exists in the 241-BX, 241-BY, 241-C, 241-AY, and 241-AZ Farms. The A1SltCk inventory estimated by the HDW model for these farms totals to 26.5 kL (7 kgal) of solids. For this reason, the A1SltCk receipt tanks in these farms were not considered for use in development of the A1SltCk waste type template.

Tank 241-A-101 received 13,975 kL (3,692 kgal) of A1SltCk concentrates from the fourth quarter of 1976 to the fourth quarter of 1980 (Agnew et al. 1996). This waste produced approximately 950 kL (251 kgal) of A1SltCk solids according to the HDW model (Agnew et al. 1997a). Two core samples (cores 154 and 156) with 19 segments each were obtained from this tank (Field et al. 1997b). These cores indicate a waste height level of 902 cm (355 in.), which is slightly higher than the inventory currently maintained in Hanlon or cited in the HDW model (953 kgal or 345 in. of waste). The high liquid inventory in segments 11 through 19 of core 154 and in segments 10 through 18 of core 156 indicate that the solids are floating on a liquid layer in this tank. While the analyte concentrations are fairly consistent as a function of depth, the liquid enriched segments have a much lower concentration of TIC and TOC. Because of this, the A1SltCk solids waste layer will be based only on the solids in segments 1 through 10 of core 154 and segments 1 through 8 of core 156. According to the HDW model, a small layer, 11.35 kL (3 kgal), of P1 waste also exists in the bottom of tank 241-A-101. This layer is characterized by high concentrations of Fe, Ni, Zr, ^{90}Sr, and ^{137}Cs, based on the composition of P2 waste in tank 241-AX-104. A review of the segment level data from segment 19 of core 154 and segment 18 of core 156 confirms that the P1 layer was not intersected by either core. The best estimate of the date of analysis for decay calculations is September 1, 1996.

Tank 241-A-102 was the primary feed tank to the 242-A Evaporator from 1976 to 1980. Although direct receipt of A1SltCk concentrate is not shown in the waste transaction records, A1SltCk waste is thought to have been accumulated in tank 241-A-102 from secondary transfers from a number of evaporator bottoms receiver tanks in the 241-A, 241-AX, 241-BX, 241-BY, 241-C, 241-AY, and 241-AZ farms. According to the HDW model, about 11.4 kL (3 kgal) of SRR and 128.6 kL (34 kgal) of A1SltCk waste are also assumed to be in this tank. The total A1SltCk layer is, therefore, only 34 cm (13.4 in.) deep. Two recent auger samples were obtained from this tank in 1995 and 1996 (Jo et al. 1996b). Because auger samples typically retrieve

48 cm (19 in.) or so of waste, the auger samples from tank 241-A-102 are assumed to represent the average composition of A1SltCk and SRR layers in this tank. The possible contribution from the SRR layer can best be judged by comparing key analytes with the A1SltCk waste in tank 241-A-101, and with the SRR waste composition from the HDW model and the P2 waste from tank 241-AX-104. Based on this comparison, certain analytes were removed because these analytes appear to be inconsistent with the likely composition of A1SltCk waste. The TIC composition of tank 241-A-102 waste appears to be too low, while the compositions of Fe, Mn, Pb, Si, total U, and Zr appear to be much higher than would be expected for A1SltCk saltcake. The remaining analytes from tank 241-A-102 were used to establish the A1SltCk Waste type template. While this selective use of analytes from the analysis of tank 241-A-102 waste may bias the results, this approach is expected to eliminate those components that were most likely added from extraneous sources, such as SRR or P2 waste. The best estimate of the date of analysis for decay calculations is April 1, 1996.

Tank 241-A-103 received 11,650 kL (3,078 kgal) of A1SltCk concentrate from the third quarter of 1976 to last quarter of 1980. According to the HDW model, this tank contains 11.35 kL (3 kgal) of AR waste and 1,374 kL (363 kgal) of A1SltCk. Two composite core samples were obtained from this tank in 1986 (Weiss and Schull 1988c). Because individual core segments were not analyzed, the composition of the AR layer can only be judged by comparing key analytes with A1SltCk from tank 241-A-101, and with the AR waste composition from the HDW model and P2 waste from tank 241-AX-104. Based on this assessment, only Ca and Si were determined to be inconsistent with A1SltCk. The Ca and Si concentrations in tank 241-A-103 were found to be 1,716 and 11,100 μg/g, respectively, compared to 309 and 369 μg/g in the A1SltCk layer in tank 241-A-101. These analytes were the only ones from tank 241-A-103 that were not included as being part of the A1SltCk layer. The best estimate of the date of analysis for decay calculation is March 1, 1987.

Tank 241-A-106 received 3,183 kL of A1SltCk concentrate from the first quarter of 1977 to the third quarter of 1980 (Agnew et al. 1996). The HDW model indicates an inventory 64.3 kL (17 kgal) of SRR waste and 79.5 kL (21 kgal) of AR under a 283.9 kL (75 kgal) layer of A1SltCk (Agnew et al. 1997b). Two composite core samples were obtained from this tank in 1986 (Weiss and Schull 1988d). Since the composite samples represent an average of all wastes in the tank, the relative contribution from wastes other than those of immediate interest must be established. The possible contributions from AR and SRR wastes were, therefore, determined by comparing key analytes with A1SltCk waste from tank 241-A-101, and with the AR and SRR compositions from the HDW model and with P2 waste from tank 241-AX-104. Based on this comparison, Ca, Fe, Mn, Pb, PO_4, Si, Zr, ^{90}Sr, and $^{239/240}Pu$ were removed from the list of analytes used to define the A1SltCk layer. The Ca and Fe values from tank 241-A-106 (5,270 and 26,350 μg/g, respectively) were much higher than the values found in the A1SltCk layer (309 and 440 μg/g) in tank 241-A-101. Manganese, Pb, PO_4, Si, Zr, and ^{90}Sr were excluded for the same reason. The best estimate of the date of analysis for decay calculations is January 1, 1988.

Tank 241-AX-103 received 3,463 kL (915 kgal) of A1SltCk concentrate from the first quarter of 1979 to the fourth quarter of 1980. Cores 212 and 214, with 2 segments each, were recovered from this tank in 1997 (Jo et al. 1998a). The first segment of core 212 was found to be empty, but satisfactory recoveries were obtained from the other segments of both cores. According to the HDW model, this tank contains 370.9 kL (98 kgal) of A1SltCk over 53 kL (14 kgal) of P2 waste. The P2 layer is assumed to be similar in composition to the P2 layer in tank 241-AX-104. Both tanks received similar volumes of P2 waste from 1965 to 1967, and both were sluiced in 1976-77, leaving a small inventory of P2 waste in tank 241-AX-104. The auger samples from tank 241-AX-104 were used to determine whether the P2 layer still exists in tank 241-AX-103. Based on the relative concentrations of Fe, Ni, Zr, ^{90}Sr, ^{137}Cs, and $^{239/240}$Pu, it appears that the P2 layer was completely removed during the 1976-77 sluicing campaign. Therefore, all recovered segments (segment 2 of core 212 and segments 1 and 2 of core 214) are assumed to be from the A1SltCk layer. The best estimate of the date of analysis for decay calculations is September 1, 1996.

The composition of the A1SltCk waste type template is based on sample values from tanks 241-A-101, 241-A-102, 241-A-103, 241-A-106, and 241-AX-103. The spreadsheet for the A1SltCk waste type template composition (A1SltCk.xls) is stored on fileserver AP001\STDINV\RPP-8847\A1SltCk.xls.

A.5.6 S1SltCk TANK WASTE LAYER ASSESSMENT

S1SltCk concentrate was produced by the 242-S Evaporator from 1972 to 1977. These concentrates were sent to a number of bottom receiver tanks, including ten tanks in the 241-S Farm, six tanks in the 241-SX Farm, eight tanks in the 241-U Farm and, finally, tank 241-SY-102. These tanks are candidates for S1SltCk layers. Five of the tanks in the 241-S Farm (241-S-103, 241-S-105, 241-S-108, 241-S-110, and 241-S-112), four tanks in the 241-SX Farm (241-SX-102, 241-SX-103, 241-SX-104, and 241-SX-105) and tank 241-U-111 were excluded from further consideration because adequate samples were not obtained from these tanks. This leaves 12 tanks for further consideration: 241-S-101, 241-S-102, 241-S-106, 241-S-107, 241-S-109. 241-S-111, 241-SX-106, 241-U-102, 241-U-103, 241-U-106, 241-U-108, and 241-U-109. Tanks 241-U-102, 241-U-105, and 241-U-107 received from 1,500 to 2,250 kL of S1SltCk concentrates, but the TLM model assigns this S1SltCk fraction to a larger amount of S2SltCk concentrate added from the 242-S Evaporator during the period from 1977 to 1980. The small amount of S1SltCk solids that would have resulted from these receipts would not be possible to distinguish from the S2SltCk. For this reason, tanks 241-U-102, 241-U-105, and 241-U-107 were excluded from the S1SltCk waste type template. Tank 241-S-107 was also deemed to be unacceptable because the S1SltCk layer is only 43 cm (17 in.) thick and is too thin to be useful.

Tank 241-S-101 received 8,350 kL (2,206 kgal) of S1SltCk concentrates from the fourth quarter of 1973 to the first quarter of 1976 (Agnew et al. 1996). This waste produced approximately 458 kL (121 kgal) of S1SltCk solids according to the HDW model (Agnew et al. 1997a). Three core samples (cores 137, 138, and 142) were recently obtained from this tank (Kruger et al.

1996). These cores show that the waste level is consistent with recent waste level measurements from the tank (about 380 cm [150 in.] of waste). Most segments from core 137 were found to be empty, except for segments 5, 7, 8, and 9. Better recoveries were obtained from cores 138 and 142. The liquid inventory in segments 1R, 2R1, and 3R suggests that a pocket of liquid exists at the center of the tank, in contrast to the wet sludge that was recovered from segments 1, 2, and 3 of core 142 at the periphery of the tank. According to the HDW model, the S2SltCk layer should be about 76.2 cm (30 in.) thick, while the S1SltCk layer is 111.2 cm (43.8 in.) thick. Based on these projections, the S1SltCk layer should be intercepted by segment 5 of core 138 and segments 3 and 4 of core 142. In this case, the layers do not appear to be horizontally arranged at the same depth for all cores. The compositions of the core segments are such that the S2SltCk and S1SltCk layers cannot be easily distinguished from one another. The S1SltCk layer, however, appears to be slightly enriched in uranium relative to the S2SltCk layer. Based on this enrichment, the segments expected to intercept S1SltCk layer were selected as the S1SltCk layer. The best estimate of the date of analysis for decay calculations is May 1, 1996.

Tank 241-S-102 was the primary feed tank for the 242-S Evaporator during the S1SltCk campaign. The evaporator records show that tank 241-S-102 did not receive any S1SltCk evaporator bottoms during the first evaporator campaign, but did receive 6,540 kL (1,728 kgal) of S2SltCk waste during the third quarter of 1980 (Agnew et al. 1996). This receipt record is not consistent with saltcake inventory profiles in the HDW model, which show that approximately 772 kL (204 kgal) of S1SltCk solids and 1,209 kL (341 kgal) of S2SltCk solids accumulated in this tank. The S1SltCk layer is expected to be represented by the lower half of segment 8 and segments 9, 10, and 11 of cores 125 and 130, excluding the small sludge layer at the bottom of the tank (Eggers et al. 1996b). The oxalate, TOC, Fe, F, and PO_4 concentrations appear to be substantially higher in segments 8, 9, and 10 than in the upper segments of cores 125 and 130, which are considered to be S2SltCk. Segments 8, 9, and 10 also have a much lower concentration of NO_3 than found in the upper segments of cores 125 and 130. These distinguishing characteristics were used to separate the S2SltCk and S1SltCk layers from one another. In this assessment, we acknowledge that the evaporator feed and receipt records are not complete, but are assuming that tank 241-S-102 received two major types of waste from the 242-S Evaporator, S1SltCk in 1975 to 1976 and S2SltCk in 1977 to 1980. This assessment is consistent with the HDW model and segment compositions observed in the core samples. For composition of the S1SltCk layer in tank 241-S-102, the average of the analytical results for segments 9 and 10 of cores 125 and 130 will be calculated. Segment 8 was excluded because of the risk of contamination from the S2SltCk layer, and segment 11 was excluded because of the sludge constituents in the segment. The average density of the S1SltCk layer is 1.61 g/mL, while the moisture content is 30.62 percent. The best estimate of the date of analysis for decay calculations is April 1996.

Tank 241-S-106 received 10,870 kL (2,672 kgal) of S1SltCk concentrates from the fourth quarter of 1973 through the first quarter of 1975 (Agnew et al. 1996). The HDW model projects an inventory of 121 kL (32 kgal) of CWR1 waste and 1,677 kL (443 kgal) S1SltCk solids. Two core samples were obtained from this tank in 1997 (Field et al. 1998b). Core 183 consisted of 10 segments, only three of which contained wet salt or salt slurry. Core 184 contained 9 segments or partial segments, three of which contained wet salt or salt slurry. The overall

waste depth cannot be judged from the core samples because these cores did not reach the bottom of the tank. Many of the segments were also found to be empty or filled with liquid. The core sample recovery profiles show that the S1SltCk layer should be represented by segments 8 and 9 of core 183 and segment 6R of core 184. The best estimate of the date of analysis for decay calculations is April 1, 1997.

Tank 241-S-109 received 8,077 kL (2,134 kgal) of S1SltCk concentrates during the fourth quarter of 1976 (Agnew et al. 1996). The HDW model projects an inventory of 49 kL (13 kgal) of CWR1 waste and 1,870 kL (494 kgal) of S1SltCk solids. Two core samples were obtained from this tank in 1996 (Field et al. 1997c). Core 158 consists of 7 segments of dry salt. This core penetrated only about 6 feet into the waste. Core 160 contained 5 segments of dry salt and liquid, but also penetrated only a short distance into the waste. The overall waste depth appears to be 557.9 cm (227.5 in.), based on the reference marks for the core samples. According to the HDW model, the waste inventory is 1,919 kL (507 kgal), which is equivalent to a depth of only 466.3 cm (183.6 in.) of waste. Hanlon (1999) reports a waste inventory of 2,100 kL (555 kgal) which is consistent with 513 cm (202 in.) of waste in the tank. Because all of the segments recovered consist of saltcake, the composition for S1SltCk layer will be taken from the statistical analysis report for this tank (Table B3-5 in Field et al. 1997c). The best estimate of the date of analysis for decay calculations is August 1, 1996.

Tank 241-S-111 received 9,637 kL (2,546 kgal) of S1SltCk concentrates from the 242-S Evaporator during the fourth quarter of 1976 (Agnew et al. 1996). Two core samples (cores 149 and 150) were taken from this tank in 1996 (Conner et al. 1997b). Core 149 was a complete core consisting of 11 segments that intercepted approximately 529.8 cm (208.6 in.) of waste. Core recoveries of 95 to 100 percent recovery were obtained for 9 segments while the lowest recovery for a segment was 68 percent. Core 150 was less successful, consisting of only three segments. Segments from core 151 were not analyzed because so few segments were successfully recovered. According to the HDW model, the inventory consists of 250 kL (66 kgal) R1 waste, 45.2 kL (12 kgal) of CWR1 waste, 230.9 kL (61 kgal) of an unknown waste, and 1,510 kL (391 kgal) of S1SltCk. Based on this inventory, the projected total waste depth should be 495 cm (194.9 in.). This compares favorably to the inventory reported by Hanlon (1999): 1,430.7 kL (378 kgal) of saltcake, 526.1 kL (139 kgal) of sludge, and 87 kL (23 kgal) of supernatant. The S1SltCk inventory predicted by the HDW model is expected to be intercepted by core segments 1 through 7. Because the first three segments of core 149 contained only liquid, the S1SltCk layer will be represented by segments 4 through 7 of this core. The best estimate of the date of analysis for decay calculations is July 1, 1996.

Tank 241-SX-106 received 24,590 kL (6,473 kgal) of S1SltCk concentrates during the fourth quarter of 1976 and 9,125 kL (2,411 kgal) of S2SltCk concentrates in the third quarter of 1978. Two core samples (core 223 and 224) were obtained from this tank in 1997 (Field et al. 1998c). Core 223 consists of 11 segments, with segment recoveries ranging from a low of 42 percent for one segment to a high of 95 to 100 percent for 5 of the segments. Most segments contained liquid with a small fraction of solids. Only segments 7 and 10 were found to have an appreciable amount of solids. Core 224 consisted of 11 segments, only 3 of which contained a significant fraction of solids (segments 8, 9, and 11). The HDW model predicts two main layers in the tank.

The S2SltCk layer is thought to contain 1,476 kL (390 kgal) of S2SltCk solids, while the underlying S1SltCk layer contains 325 kL (86 kgal) of S1SltCk solids. Based on these figures, the S1SltCk layer would constitute the solids in the bottom 79 cm (31.1 in.) of waste, that is, the solids in segments 10 and 11 of cores 223 and 224. Segment 11 of core 223 was found to be empty, as was segment 10 of core 224. The only segments with useable samples from the S1SltCk layer were segment 10 from core 223 and segment 11 from core 224. These samples were used to generate the S1SltCk layer composition for tank 241-SX-106. There are minor discrepancies in the volume of waste reported for this tank. The January 1, 1994 measured waste depth was 512 cm (201.6 in.) which corresponds to an inventory of 2,044 kL (540 kgal). The inventory reported by Hanlon for January 1, 1994 was 2,037 kL (538 kgal), consisting of 231 kL (61 kgal) supernatant, 45 kL (12 kgal) of sludge, and 1,760 kL (465 kgal) of saltcake. The HDW model predicts 1,806 kL (477 kgal) of saltcake. In November-December 1997, when the tank was sampled, the waste height had decreased to 502 cm (197.7 in.) which corresponds to an inventory of 2,003 kL (529 kgal). The best estimate of the date of analysis for decay calculations is January 1, 1998.

Tank 241-U-103 received 2,252 kL (595 kgal) of S1SltCk concentrates in the first quarter of 1977 and 2,199 kL (581 kgal) of S2SltCk concentrates during the second quarter of 1977. The HDW model projects an inventory of 121 kL (32 kgal) of MW, 1,143 kL (302 kgal) of S1SltCk solids, and 458 kL (121 kgal) of S2SltCk solids. Three core samples were taken from tank 241-U-103 (cores 176, 182, and 175). Core 176 consisted of 9 segments, 5 of which were empty. Segment 6 is the only one that contained appreciable solids. Core 182 consisted of 10 segments, only one of which was empty. Core segment recoveries varied from a low of 58 percent for segment 10 to 95 to 100 percent for 4 other segments. For core 182, the measured waste depth was 449 cm (177.1 in.). The waste volume cited by the HDW model corresponds with a waste depth of 418.6 cm (164.8 in.). Hanlon (1999) reports a waste inventory of 49 kL (13 kgal) supernatant, 121 kL (32 kgal) sludge, and 1,601 kL (423 kgal) saltcake. The total is 1,771 kL (468 kgal) or 431 cm (169.5 in.) of waste. The third core, core 175, contained 3 segments or partial segments. Based on the projected location of the S1SltCk layer, segments 4 through 8 of core 176 and segments 4 through 9 of core 182 intercepted the S1SltCk layer. Because certain segments were empty or contain only liquid, the useable segments from the S1SltCk layer were segment 5 from core 176 and segments 5, 6, and 7 of core 182. The best estimate of the date of analysis for decay calculations is January 1, 1998.

Tank 241-U-106 received 2,267 kL (599 kgal) of S1SltCk concentrates during the first quarter of 1977 and only 30 kL (8 kgal) of S2SltCk concentrates during the second quarter of that year. The HDW model shows an inventory of 98.4 kL (26 kgal) of MW and 700 kL (185 kgal) of S1SltCk solids. The waste transaction records, however, show that metal waste was sluiced out of this tank between December 1956 and January 1957. In January of that year, tank 241-U-106 was inspected and found to be empty (Rhodenhizer 1987). The waste transaction records also show that some 98.4 kL (26 kgal) of metal waste was subsequently added to tank 241-U-106 during the third quarter of 1960, about four years after T-Plant was shut down. This transfer probably consisted of residual metal waste, but it is unlikely that this slurry would have produced the volume of MW solids (98.4 kL [26 kgal]) predicted in tank 241-U-106. This tank then received REDOX waste supernatant from tank 241-SX-103 via tank 241-U-101 during the same quarter and a final transfer of additional supernatant from tank 241-C-104 in 1974. These were

the last transfers before the 242-S Evaporator campaign in 1975-1976, during which time a substantial volume of S1SltCk concentrates were added to this tank. Two core samples (cores 147 and 148) were taken from this tank in May 1996 (Brown et al. 1997). Each core consists of 5 segments, which corresponds to a projected inventory of 992 kL (262 kgal) of saltcake, sludge, and supernatant. Hanlon (1999) reports an inventory of 56.7 kL (15 kgal) of supernatant and 799 kL (211 kgal) of saltcake for a total of 855 kL (226 kgal) of waste. Based on the estimates in HDW model, the S1SltCk layer is located in segments 2 through the upper half of segment 5, while the MW layer is in the lower half of segment 5. The concentrations for Al, Bi, Fe, Ni, and U were found to be uniform for all segments, without any indication of enrichment or depletion in the bottom segment (segment 5). For NO_3, NO_2, PO_4, and SO_4, the results were quite different, with definite signs of NO_3 and PO_4 enrichment and NO_2 and SO_4 depletion in segment 5. These features are consistent with a MW layer in segment 5 at the bottom of the tank. Based on the large inventory of liquid in segments 1 and 2 and the possibility of a sludge layer in segment 5, the S1SltCk layer most likely resides in segments 3 and 4. These segments were used to construct the S1SltCk waste type template. Because tank 241-U-106 received complexed waste from the 242-S Evaporator after 1978, the high TOC value reported for this waste will not be used in the S1SltCk waste type template (Reynolds 1998). The best estimate of the date of analysis for decay calculations is January 1, 1996.

Tank 241-U-108 received 3,198 kL (855 kgal) of S1SltCk concentrates during the first quarter of 1977. According to the HDW model, this tank contains 11.3 kL (3 kgal) of MW, 90.8 kL (24 kgal) of CWR2, 874.3 kL (231 kgal) of S1SltCk, and 696.4 kL (184 kgal) of S2SltCk for a total of 1,672.8 kL (442 kgal) waste. Metal waste was sluiced out of this tank in 1953, and once again in 1956, with the tank being inspected and declared empty in February 1957 (Rhodenhizer 1987). The S2SltCk layer predicted by the HDW model is not consistent with the *Waste Status and Transaction Record Summary* (Agnew et al. 1997b), which indicates that only 374.7 kL (99 kgal) of S2SltCk concentrates were sent to tank 241-U-108 during 1977. Three core samples (cores 141, 145, and 146) were obtained from this tank in 1996 (Bell and Cromar 1997). Each core consisted of 9 segments, representing a total inventory of about 1,783 kL (471 kgal) or 434 cm (170.7 in.) of waste. Hanlon (1999) reports the inventory as 91 kL (24 kgal) of supernatant, 110 kL (29 kgal) of sludge and 1,571 kL (415 kgal) of saltcake for a total of 1,771 kL (468 kgal) of waste. Based on the predictions of the HDW model, the S1SltCk layer at the bottom of the tank was intercepted by the lower half of segment 5 through segment 9. A review of the NO_3 concentrations of the segments indicates that segments 2 through the upper half of segment 5 contain higher concentrations of NO_3 than the lower segments, suggesting a possible juncture between the S1SltCk and S2SltCk layers. A somewhat stronger juncture appears to exist at the bottom of the tank where the lower half of segment 9 contains less Al and SO_4 but substantially more PO_4 than the upper segments. This suggests the presence of a small MW layer in this region of the tank. The Al concentrations, however, do not exhibit the high Al concentrations consistent with the CWR1 layer alleged by the HDW model to be in the bottom of the tank. Because a substantial amount of liquid exists in the first segment and S2SltCk solids presumably exist in segments 2 through 5, segments 6 through 8 are considered to be the most representative segments for the S1SltCk layer. The best estimate of the date of analysis for decay calculations is June 1996.

Tank 241-U-109 received 2,994 kL (791 kgal) of S1SltCk concentrates from 1975 and 1977 and 11.35 kL (3 kgal) of S2SltCk concentrates from tank 241-SY-102 during the second quarter of that year (Agnew et al. 1996). The HDW model projects an inventory of 90.8 kL (24 kgal) of MW, 90.8 kL (24 kgal) of secondary CWR1, 814 kL (215 kgal) of S1SltCk, and 685 kL (181 kgal) of S2SltCk. The S2SltCk estimate appears inconsistent with the volume of S2SltCk concentrates added to this tank (Agnew et al. 1997b). Three core samples (cores 123, 124, and 128) were obtained from this tank in 1996 (Baldwin and Stephans 1996). Each core contained 9 segments, representing about 1,744 kL (461 kgal) or 424 cm (166.9 in.) of waste. This inventory is a little higher than the inventory projected by the HDW model (1,680 kL of waste). Hanlon (1999) reports an inventory of 71.9 kL (19 kgal) of supernatant, 182 kL (48 kgal) of sludge, and 1,499 kL (396 kgal) of saltcake for a total of 1,752 kL (463 kgal) of waste. Based on the HDW model, the S2SltCk layer should be intercepted by segments 1, 2, 3 and the upper half of segment 4, while the S1SltCk layer should be found in the lower half of segment 4 through the upper half of segment 8. The CWR1 and MW layers should reside in segment 9. The Al and PO_4 concentrations in segment 9 are consistent with the aluminum-rich CWR1 waste and metal waste that are expected at the bottom of the tank. Segment 9 also contains less NO_3 than is found in other segments of the cores. The TIC and TOC concentrations suggest a boundary between S1SltCk and S2SltCk layers between segments 5 and 6. The upper segments (segments 1 through 5) exhibit somewhat lower concentrations of TOC and TIC than do the lower segments of the cores. While the concentrations for most of the components show very little difference across the segments, the lower segments, segments 6 through 8, will be used to represent the S1SltCk layer for this tank. The best estimate of the date of analysis for decay calculations is March 1, 1996.

The composition of the S1SltCk waste type template is based on sample values from tanks 241-S-101, 241-S-102, 241-S-106, 241-S-109, 241-S-111, 241-SX-106, 241-U-103, 241-U-106, 241-U-108, and 241-U-109. The spreadsheet for the S1SltCk waste type template composition (S1SltCk.xls) is stored on fileserver AP001\STDINV\RPP-8847\S1SltCk.xls.

A.5.7 S2SltCk TANK WASTE LAYER ASSESSMENT

S2SltCk waste was produced by the 242-S Evaporator from 1977 to 1980. This waste was routed to 21 different receiving tanks in the 241-A, 241-S, 241-SX, 241-SY, and 241-U Farms (Agnew et al. 1996). These tanks collectively received about 21,000 kL (5,500 kgal) of S2SltCk concentrate and contain 8,950 kL (2,364 kgal) of S2SltCk solids, with about 18 percent of this inventory residing in tanks that have not been sampled in the 241-S, 241-SX, 241-SY, and 241-U Farms. Eight of these tanks were selected for candidate S2SltCk waste layers, including tanks 241-S-101, 241-S-102, 241-SX-106, 241-SY-103, 241-U-102, 241-U-103, 241-U-105, and 241-U-107. The S2SltCk layer in tank 241-S-107 is too thin (20 cm or 8 in.) to be realistically included in the S2SltCk waste type template. Several other tanks were rejected because of apparent inconsistencies in the HDW model. Two of these tanks are listed as having received S2SltCk waste but no longer contain this waste (tanks 241-A-101 and 241-S-110), while another tank (tank 241-U-108) evidently contains this waste but was not listed as one of the original receivers. One other tank, 241-U-109, is listed as having received only a very small amount of

S2SltCk waste but the HDW model projects a substantial inventory of this waste. Eight other tanks also received S2SltCk waste but have not been sampled or, for various reasons, the samples were found to be not usable for the S2SltCk waste type template (tanks 241-S-103, 241-SX-101, 241-SX-102, 241-SX-103, 241-SX-104, 241-SX-10, 241-SY-101, and 241-U-106).

Tank 241-S-101 received 363 kL (96 kgal) of S2SltCk concentrates from the third quarter of 1977 to the first quarter of 1980 (Agnew et al. 1996). This waste receipt produced approximately 83 kL (22 kgal) of S2SltCk solids according to the HDW model (Agnew et al. 1997a). Three core samples (cores 137, 138, and 142) were obtained from this tank (Kruger et al. 1996). These cores show that the waste level is consistent with recent waste level measurements from the tank (about 150 inches of waste). Most segments from core 137 were found to be empty, except for segments 5, 7, 8, and 9. Better recoveries were obtained from cores 138 and 142. The liquid inventory in segments 1R, 2R1, and 3R suggest that a pocket of liquid exists at the center of the tank, in contrast to the wet sludge that was recovered from segments 1, 2, and 3 of core 142 at the periphery of the tank. According to the HDW model, the S2SltCk layer should be about 76.2 cm (30 in.) thick, while the S1SltCk layer is 111.2 cm (43.8 in.) thick. Based on these projections, the S2SltCk layer should be intersected by segments 1 and 2 of core 142 and the S1SltCk layer by segment 5 of core 138 and segments 3 and 4 of core 142. In this case, the layers do not appear to be horizontally arranged at the same depth for all cores. The compositions of the core segments are such that the S2SltCk and S1SltCk layers cannot be easily distinguished from one another. The S1SltCk layer, however, appears to be slightly enriched in uranium relative to the S2SltCk layer. Based on this enrichment, the segments expected to intercept S2SltCk layer were selected as the S2SltCk layer. The best estimate of the date of analysis for decay calculations is April 1996.

Evaporator records show that tank 241-S-102 did not receive any S1SltCk evaporator bottoms during the first evaporator campaign, but did receive 6,540 kL (1,728 kgal) of S2SltCk during the third quarter of 1980 (Agnew et al. 1996). This receipt record is not consistent with saltcake inventory profiles in the HDW model, which show that approximately 772 kL (204 kgal) of S1SltCk solids and 1,209 kL (341 kgal) of S2SltCk solids accumulated in this tank. The S2SltCk layer should be represented by segments 2 through 7 of cores 125 and 130 (Eggers et al. 1996b). The oxalate, TOC, Fe, F, and PO_4 concentrations appear to be substantially higher in segments 8, 9, and 10 than in the upper segments of cores 125 and 130. Segments 8, 9, and 10 also have a much lower concentration of NO_3 than found in the upper segments of cores 125 and 130. These distinguishing characteristics were used to distinguish the S2SltCk and S1SltCk layers from one another. In this assessment, we acknowledge that the evaporator feed and receipt records are not complete, but are assuming that tank 241-S-102 received two major types of waste from the 242-S Evaporator, S1SltCk in 1975 to 1976 and S2SltCk in 1977 to 1980. This assessment is consistent with the HDW model and segment compositions observed in the core samples. The best estimate of the date of analysis for decay calculations is April 1996.

Tank 241-SX-106 received 2,411 kL (637 kgal) of S2SltCk concentrates during the first quarter of 1978 to the fourth quarter of 1980. Two core samples (cores 223 and 224) were obtained from this tank in 1997 (Field et al. 1998c). Core 223 consists of 11 segments, with segment recoveries ranging from a low of 42 percent for one segment to 95 to 100 percent for 5 other segments.

Most segments contained liquid with a small fraction of solids. Only segments 7 and 10 were found to have an appreciable amount of solids. Core 224 also was comprised of 11 segments, only 3 of which contained a significant fraction of solids (segments 8, 9, and 11). There are minor discrepancies in the volume of waste reported for this tank. The HDW model predicts two main layers in the tank. The S2SltCk layer is thought to contain 1,476 kL (390 kgal) of S2SltCk solids, while the underlying S1SltCk layer contains 325 kL (86 kgal) of S1SltCk solids for a total of 1,806 kL (477 kgal). The January 1, 1994 measured waste depth of 512 cm (201.6 in.) corresponds to an inventory of 2,037 kL (538 kgal). The inventory reported by Hanlon for January 1, 1994 was 2,037 kL (538 kgal), consisting of 231 kL (61 kgal) supernatant, 45 kL (12 kgal) of sludge, and 1,760 kL (465 kgal) of saltcake. In November-December 1997, when the tank was sampled, the waste height had decreased to 502 cm (197.7 in.), which corresponds to an inventory of 2,003 kL (529 kgal). Based on these figures, the S2SltCk layer should be intercepted by segments 6, 7, and 8 of core 224 and by segments 5, 6, 8, and 9 of core 224. The upper 5 segments of core 223 mostly contain liquid, while segment 6 of that core was found to be empty. Segments 2 and 4 of core 224 were also empty, while segments 1 and 3 contained mostly liquid. The S2SltCk layer is comprised of all segments between the liquid layer at the top and the S1SltCk segments at the bottom of the tank. The best estimate of the date of analysis for decay calculations is January 1, 1998.

Tank 241-SY-103 received some 2,271 kL (600 kgal) of S2SltCk concentrate from the first quarter of 1978 to the last quarter of 1980. According to the HDW model, 2,184 kL (577 kgal) of S2SltCk solids exist in this tank. One core (core 62) was obtained from this tank in 1994 (Hanson et al. 1996). This core consists of 15 segments, all but 6 of which primarily contain liquid. Segments 7, 8, 9, 10, 11, and 12 contained solids that represent S2SltCk waste. The S2SltCk composition was derived from the average composition of the nonconvective layer in Table 4-2 of the TCR (Hansen et al. 1996). Based on the HDW model, tank 241-SY-103 is assumed to have only S2SltCk solids. The best estimate of the date of analysis for decay calculations is October 1, 1994.

Several cores were obtained from tank 241-U-102. This tank received 659 kL (174 kgal) of S2SltCk concentrates from the second quarter of 1977 to the fourth quarter of that year. This tank currently has a projected inventory 162 kL (43 kgal) of metal waste (or REDOX waste), 810 kL (214 kgal) of T2SltCk solids, and 374 kL (99 kgal) of S2SltCk solids (Agnew et al. 1996 and 1997a). The two 1996 cores from this tank (cores 143 and 144) contain a mixture of soft sludge and moist saltcake, with two distinct layers (Hu et al. 1997). Because the cores did not penetrate the sludge layer expected at the bottom of the tank, these cores are believed to represent only the two uppermost saltcake layers in the tank. The S2SltCk layer can be distinguished from the lower T2SltCk layer by the PO_4 and SO_4 concentrations. On average, the T2SltCk layer contains about 5,030 μg/g PO_4 and 15,560 μg/g SO_4, while the S2SltCk layer has 24,850 μg/g PO_4 and 12,370 μg/g SO_4. With respect to other analytes, including Na, NO_2, and NO_3, the composition of T2SltCk and S2SltCk are sufficiently similar that adjacent layers cannot be distinguished from each other. Segments 1 and 2 are believed to represent the S2SltCk layer and segments 3, 4, 5, and 6, together with sub segments 5A, 5B and 6A, the T2SltCk layer. The best estimate of the date of analysis for decay calculations is May 1996.

Tank 241-U-103 received 2,199 kL (581 kgal) of S2SltCk concentrates from the second to the fourth quarter of 1977. The HDW model projects an inventory of 121 kL (32 kgal) of MW, 1,143 kL (302 kgal) of S1SltCk solids, and 458 kL (121 kgal) of S2SltCk solids. Three core samples (cores 176, 182, and 175) were taken from tank 241-U-103 (Sasaki et al. 1998). Core 176 is comprised of 9 segments, 5 of which were empty. Segment 6 is the only one that contained appreciable solids. Core 182 consisted of 10 segments, only one of which was empty. Core segment recoveries varied from a low of 58 percent for segment 10 to 95 to 100 percent for 4 other segments. For core 182, the measured waste depth was 449 cm (177.1 in.). The waste volume cited by the HDW model is consistent with a waste depth of 418.6 cm (164.8 in.). Hanlon (1999) reports a waste inventory of 49 kL (13 kgal) supernatant, 121 kL (32 kgal) sludge, and 1,601 kL (423 kgal) saltcake. The total is 1,771 kL (468 kgal) or 431 cm (169.5 in.) of waste. The third core, core 175, contained 3 segments or fractions of segments. Based on the projected location of the SMMS2 layer, segments 1 and 2 of core 175, segment 1 of core 176, and segments 1 and 2 of core 182 intercepted the SMMS2 layer. These segments were used to construct the S2SltCk layer. The best estimate of the date of analysis for decay calculations is May 1, 1997.

Tank 241-U-105 received 1,786 kL (472 kgal) of S2SltCk concentrates between the second and fourth quarters of 1978 (Agnew et al. 1996). According to the HDW model, this tank contains 121 kL (32 kgal) of metal waste, 280 kL (74 kgal) of T2SltCk solids, and 1,040 kL (275 kgal) of S2SltCk solids (Agnew et al. 1997a). Three complete core samples (cores 131, 133, and 138) were obtained from this tank (Brown and Franklin 1996). These cores indicate the waste level is consistent with recent waste level measurements for the tank (411 cm [162 in.] of waste). The first segment is comprised of liquid, while segments 2 through 7 consist of a mixture of soft saltcake and sludge (S2SltCk waste), and segments 8 and the upper half of segment 9 consist of soft saltcake and sludge (T2SltCk layer). The lower half of segment 9 supposedly consists of soft sludge, possibly metal waste. The segment compositions, however, support a different assessment. The bottom half of segment 9 clearly consists of sludge based on the concentrations of Al, Fe, Mg, Mn, PO_4 and, especially, U. The concentrations of Al, PO_4, and U do not match with a MW layer. In the upper segments (segments 1 through the upper half of segment 9), the segment concentrations for Al, Cl, PO_4, SO_4, ^{154}Eu, and total alpha do not indicate a distinct boundary between the S2SltCk and T2SltCk saltcake layers. These two layers appear to be essentially identical to one another. Based on the chronological ordering of the saltcake layers, segments 1 through 5 of core 131, segments 2 through 4 of core 133, and segments 3 through 5 of core 136 were identified as the S2SltCk waste layer. While most upper segments appear to be a mixture of saltcake and sludge, the waste transaction records clearly support the development of saltcake layers from the 242-T and 242-S Evaporator concentrates. Because tank 241-U-105 received high strontium concentration waste from the 242-S Evaporator in 1977, the high TOC value reported for this waste will not be used in the S1SltCk waste type template (Reynolds 1998). The best estimate of the date of analysis for decay calculations is April 1, 1996.

Tank 241-U-107 received 10,810 kL (2,856 kgal) of S2SltCk concentrates from the second quarter of 1979 to the fourth quarter of 1980. This waste receipt is thought to have produced about 1,079 kL (285 kgal) of S2SltCk solids. In addition to these solids, tank 241-U-107 is also assumed to contain, from top to bottom, 53 kL (14 kgal) of (T2SltCk), and 287.6 kL (76 kgal) of

CWR1 waste. Three core samples (cores 129, 134, and 135) were taken from this tank in 1996 (Jo et al. 1996c). Sample recoveries were limited such that only four useful segments were recovered from core 129, six useful segments from core 134, and only two segments from core 135. These segments are from the top 275 cm (108 in.) of waste, which corresponds to the S2SltCk layer estimated by the HDW model. Careful examination of the segment compositions, however, reveals that segments 1, 2, and 3 are substantially different from segments 4 through 6 (with segment 4 being a transition segment between these wastes). Segments 4 through 6 have substantially higher concentrations of Al, Fe, Mn, Ni, Pb, PO_4, Si, TOC, and total alpha, but a much lower concentration of SO_4. High Al, Ni, Pb, and Si would be consistent with CWR1 waste. High Fe and PO_4 may be due to wastes of unknown origin. Because of these characteristics, only segments 2, 3, and 3R of core 129 and segments 2 and 3 of core 134 were used to represent the S2SltCk waste layer. The best estimate of the date of analysis for decay calculations is May 1, 1996.

Tank 241-U-109 received 2,994 kL (791 kgal) of S1SltCk concentrate from 1975 and 1977 from tank 241-S-102 and only 11.35 kL (3 kgal) of S2SltCk concentrate from tank 241-SY-102 during the second quarter of 1977 (Agnew et al. 1996). The HDW model, however, projects an inventory of 90.8 kL (24 kgal) of MW, 90.8 kL (24 kgal) of secondary CWR1, 813.7 kL (215 kgal) of S1SltCk, and 685 kL (181 kgal) of S2SltCk. The S2SltCk estimate is not consistent with the volume of S2SltCk concentrates added to this tank (Agnew et al. 1997b). Because of this discrepancy, tank 241-U-109 will not be used in the S2SltCk tank waste layer composition.

The composition of the S2SltCk tank waste layer is based on sample values from tanks 241-S-101, 241-S-102, 241-SX-106, 241-SY-103, 241-U-102, 241-U-103, 241-U-105, and 241-U-107. The spreadsheet for the S2SltCk waste type template composition (S2SltCk.xls) is stored on fileserver AP001\STDINV\RPP-8847\S2SltCk.xls.

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APPENDIX B

WASTE COMPOSITION SPREADSHEETS

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APPENDIX B

WASTE COMPOSITION SPREADSHEETS

The spreadsheets used to calculate the sample based composition of the identified waste types are maintained on the AP001 fileserver at STDINV\RPP-8847*.xls.

Waste Type Definition	Spreadsheet Name
$BiPO_4$ Waste Composition Spreadsheets	
$BiPO_4$ process metal waste (1944-1956)	MW.xls
$BiPO_4$ First cycle decontamination waste (1944-1956)	1C.xls
$BiPO_4$ Second cycle decontamination waste (1944-1956)	2C.xls
Lanthanum fluoride process "224" waste (1948-1956)	224.xls
Uranium Recovery and Scavenging Waste Composition Spreadsheets	
Tributyl phosphate process waste (1952-1958)	TBP.xls
Ferrocyanide sludge from in-farm scavenging of 1C supernatants in TY Farm (1955-1958)	1CFeCN.xls
Ferrocyanide sludge from in-plant scavenged supernatant (1954-1958)	PFeCN.xls
Ferrocyanide sludge from supernatants scavenged in CR Vault (1955-1958)	TFeCN.xls
REDOX Waste Composition Spreadsheets	
REDOX cladding waste, aluminum clad fuel (1952-1960)	CWR1.xls
REDOX high-level waste (1952-1958)	R1.xls
Saltcake from self-concentration in S and SX Farms (1952-1966)	RSltCk.xls
PUREX Waste Composition Spreadsheets	
PUREX high-level waste (1963-1967)	P2.xls
PUREX cladding waste (1956-1960)	CWP1.xls
PUREX cladding waste (1961-1972)	CWP2.xls
Atmospheric-Pressure Evaporation Salt Cake Composition Spreadsheets	
Saltcake from 242-B Evaporator operation (1951-1953)	BSltCk.xls
Saltcake from in-tank solidification (ITS) in BY Farm (1965-1974)	BYSltCk.xls
Saltcake from the 242-T Evaporator (1951-1955)	T1SltCk.xls
Saltcake from the last 242-T Evaporator campaign (1965-1976)	T2SltCk.xls
Vacuum Evaporator/Crystallizer Salt Cake Composition Spreadsheets	
Saltcake from the first 242-A Evaporator campaign (1977-1980)	A1SltCk.xls
Saltcake from the first 242-S Evaporator campaign using S-102 feed tank (1973-1976)	S1SltCk.xls
Saltcake from the second 242-S Evaporator campaign using SY-102 feed tank (1977-1980)	S2SltCk.xls

The spreadsheets used to calculate the vectors from the merged sample and HDW model data are stored on AP001 fileserver at STDINV\BBI\Waste Template FY01\Solid Template Version 10C.xls and Liquid Template Version 10C.xls.