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Annual Report of Tank Waste Treatability

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ANNUAL REPORT OF TANK WASTE TREATABILITY

S. A. Barker
A. G. Lane

ABSTRACT

This report has been prepared as part of the Hanford Federal Facility Agreement and Consent Order* *(Tri-Party Agreement) and constitutes completion of Tri-Party Agreement milestone M-04-00C for fiscal year 1992. This report provides a summary of treatment activities for newly generated waste, existing double-shell tank waste, and existing single-shell tank waste, as well as a summary of grout disposal feasibility, glass disposal feasibility, alternate methods for disposal, and safety issues which may impact the treatment and disposal of existing defense nuclear wastes.*

This report is an update of the 1991 report and is intended to provide traceability for the documentation of the areas listed above by statusing the studies, activities, and issues which occurred in these areas over the period of March 1, 1991, through February 29, 1992. Therefore, ongoing studies, activities, and issues which were documented in the previous (1991) report are addressed in this (1992) report.

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**Hanford Federal Facility Agreement and Consent Order*, 2 Vols., as amended, Washington State Department of Ecology, U.S. Environmental Protection Agency, and U.S. Department of Energy, Olympia, Washington.

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LIST OF TERMS

CC	complexant concentrate (waste)
CRW	cladding removal waste
CVS	composition variability study
DOE	U.S. Department of Energy
DOE-HQ	U.S. Department of Energy-Headquarters
DST	double-shell tank
DSS	double-shell slurry
DSSF	double-shell slurry feed
Ecology	Washington State Department of Ecology
EIS	Environmental Impact Statement
EM	(Office of) Environmental Management
EPA	U.S. Environmental Protection Agency
FSAR	Final Safety Analysis Report
FY	fiscal year
GTF	Grout Treatment Facility
HLW	high-level waste
IPM	Initial Pretreatment Module
HWVP	Hanford Waste Vitrification Plant
LFL	lower flammability limit
LLW	low-level waste
MFT	melter feed tank
NCAW	neutralized current acid waste
NCRW	neutralized cladding removal waste
NEPA	*National Environmental Policy Act of 1969*
OCRWM	Office of Civilian Radioactive Waste Management
ORNL	Oak Ridge National Laboratory
PA	Performance Assessment
PFP	Plutonium Finishing Plant
PNL	Pacific Northwest Laboratory
PRF	Plutonium Reclamation Facility
PSW	phosphate/sulfate waste
PUREX	Plutonium/Uranium Extraction (Plant)
RL	U.S. Department of Energy, Richland Field Office
SAR	Safety Analysis Report
SEIS	supplemental environmental impact statement
SME	slurry mix evaporator
SRAT	slurry receipt and adjustment tank
SREX	strontium extraction
SRL	Savannah River Laboratory
SST	single-shell tank
Tri-Party Agreement	*Hanford Federal Facility Agreement and Consent Order*
TRU	transuranic (waste)
TRUEX	transuranium extraction
TWRS	Tank Waste Remediation System
U.S.	United States
UST/ID	Underground Storage Tank/Integrated Demonstration
WAC	*Washington Administrative Code*
WAPS	Waste Acceptance Preliminary Specifications
Westinghouse Hanford	Westinghouse Hanford Company

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ANNUAL REPORT OF TANK WASTE TREATABILITY

1.0 INTRODUCTION

1.1 TRI-PARTY AGREEMENT

The *Hanford Federal Facility Agreement and Consent Order* (Tri-Party Agreement) (Ecology et al. 1990), established in 1989 by the U.S. Department of Energy (DOE), the U.S. Environmental Protection Agency (EPA), and the Washington State Department of Ecology (Ecology), provides the basis for this report. The Tri-Party Agreement contains milestone M-04-00, which addresses tank waste treatability, issues, and concerns.

Milestone M-04-00 requires that reports of tank waste treatability studies be submitted annually beginning in September 1990.

1.2 MILESTONE M-04-00A, ANNUAL TANK WASTE TREATABILITY 1990 REPORT

The 1990 *Annual Report of Tank Waste Treatability* (Karnesky 1990) documented the first of an annual series of reports required by milestone M-04-00. In addition to presenting an historical perspective of tank waste treatment at the Hanford Site, this report described planned treatment of existing double-shell tank (DST) and single-shell tank (SST) wastes, and provided the technical basis for selection of grout and borosilicate glass as disposal forms.

1.3 MILESTONE M-04-00B, ANNUAL TANK WASTE TREATABILITY 1991 REPORT

The 1991 report (Giese 1991) represented the first statusing report in the series of these annual reports. The organization of the 1991 report was the same as that of the 1990 version. Two additional sections were added to the 1991 report. Section 7.0 summarized alternative treatment/disposal technologies which could have an impact on future disposal. Section 8.0 contained pertinent issues which may affect either treatability of tank waste or the feasibility of using grout or glass (or another viable alternative) as a final disposal option.

1.4 MILESTONE M-04-00C, ANNUAL TANK WASTE TREATABILITY 1992 REPORT

The 1992 *Annual Report of Tank Waste Treatability* also follows organization of the previous reports, comprising the second statusing report in this series of milestone reports.

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2.0 SUMMARY

This third annual report satisfies the Tri-Party Agreement milestone M-04-00C for fiscal year (FY) 1992.

2.1 DOUBLE-SHELL TANKS

Existing waste in ten DSTs will be pretreated to separate the waste into high-level waste (HLW), transuranic (TRU) waste, and low-level waste (LLW) volumes. Eighteen DSTs are currently designated as LLW and are planned to be transferred directly to grout disposal.

Treatment of the separated HLW and TRU waste fractions will consist of vitrification in the Hanford Waste Vitrification Plant (HWVP) before disposal in a geologic repository. Treatment of the LLW consists of solidification in cement-based grout before disposal in near-surface vaults at the Hanford Site. These treatment processes are in various stages of development and are discussed in Section 3.0 on DST waste treatability.

2.2 SINGLE-SHELL TANKS

Existing waste in SSTs continues to be characterized to enable appropriate treatment options to be developed. This information is needed for a supplemental environmental impact statement (SEIS) leading to a decision on final SST waste disposal.

Studies which address treatment and disposal options were performed in FY 1991. Some of these are ongoing activities which are revised as new information becomes available and will be incorporated in a subsequent report.

2.3 GROUT AND GLASS

The current grout treatment process for LLW is described in Section 3.8. Major processing requirements for waste vitrification of the HLW in HWVP are also discussed in Section 3.8.

2.4 CURRENT WASTE GENERATORS

Currently, the following ten major facilities generate waste subject to this study report.

- 100-N Area
- 300 Area
- 400 Area
- Tank farms
- Evaporators
- Plutonium Finishing Plant (PFP)
- Plutonium/Uranium Extraction (PUREX) Plant
- B Plant

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- S Plant
- T Plant.

Treatment of these wastes are addressed in Appendix A.

2.5 UNREVIEWED SAFETY QUESTIONS

This section contains pertinent issues which may affect either the treatability of tank waste or the feasibility of using glass or grout (or another viable alternative) as a final disposal option.

The five major issues that are summarized in the 1992 report are:

- Hydrogen issue
- Ferrocyanide issue
- Organic issue
- High-heat tanks issue
- Criticality issue.

2.6 ALTERNATIVE TREATMENT/DISPOSAL TECHNOLOGIES

This section summarizes alternative treatment/disposal technologies which may have an impact on future disposal.

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3.0 TREATMENT OF EXISTING DOUBLE-SHELL TANK WASTES

This section documents the studies, activities, and issues that occurred in this area over the period of March 1, 1991, through February 29, 1992.

3.1 INTRODUCTION

Treatment of existing DST wastes is required before permanent disposal (Augustine 1989). The treatment strategy is to separate DST wastes into three portions: HLW, TRU waste, and LLW. Ten DSTs will be pretreated to separate the waste into HLW, LLW, and TRU volumes. Eighteen DSTs are currently designated as LLW and are planned to be sent directly to grout disposal.

Treatment of the separated HLW and TRU waste fractions will consist of vitrification in the HWVP before disposal in a Federal geologic repository. Treatment of the LLW consists of solidification in cement-based grout before disposal in near-surface vaults at the Hanford Site.

These treatment processes are in various stages of development as discussed below. The planned treatment activities will be discussed according to the waste types of double-shell slurry feed (DSSF), double-shell slurry (DSS), neutralized current acid waste (NCAW), neutralized cladding removal waste (NCRW), PFP waste, and complexant concentrate (CC) waste.

The current waste volume inventory of the Hanford Site tank farms as of February 1992 is listed in Table 3-1. This information is available from the *Tank Farm Surveillance and Waste Status Summary Report for February 1992*, WHC-EP-0182-47 (Hanlon 1992). The volumes of both solids and liquids are recorded in thousands of gallons.

Tables 3-1 and 3-2 contain references to designations for waste types other than NCAW (designated as aging), NCRW (designated PN/PD), PFP (designated PT), CC, DSS, and DSSF. The concentrated phosphate (designated CP) waste is currently planned to be grouted directly. The dilute complexed (designated DC) waste will become CC waste and the dilute noncomplexed (designated DN) will become DSS/DSSF after concentration.

3.2 PLANNED TREATMENT OF DOUBLE-SHELL SLURRY FEED AND DOUBLE-SHELL SLURRY

3.2.1 Definition of Double-Shell Slurry Feed and Double-Shell Slurry

Many streams that enter DSTs consist of dilute liquids low in radioactivity. These streams are so concentrated by Evaporator 242-A that a second pass through the 242-A Evaporator would increase the sodium aluminate concentration past the sodium phase boundary, and the stream would solidify when cooled. At this point the waste is called DSSF. When the DSSF is processed through Evaporator 242-A, the DSSF is concentrated past the sodium

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Table 3-1. Double-Shell Tank Inventory as of February 1992. (2 sheets)

Tank Number	Waste material[a]	Volume in kgal (m^3)				
		Total waste	Supernatant[b]	DSS	Sludge	Salt cake
101-AN	DN	628 (2,377)	628 (2,377)	0	0	0
102-AN	CC	1,094 (4,141)	1,005 (3,804)	0	89 (337)	0
103-AN	DSS	949 (3,592)	12 (45)	937 (3,547)	0	0
104-AN	DSSF	1,064 (4,027)	800 (3,028)	0	264 (999)	0
105-AN	DSSF	1,129 (4,273)	1,129 (4,273)	0	0	0
106-AN	CP	1,015 (3,842)	998 (3,777)	0	17 (64)	0
107-AN	CC	1,074 (4,065)	940 (3,558)	0	134 (507)	0
101-AP	DN	1,062 (4,020)	1,062 (4,020)	0	0	0
102-AP	DN	133 (503)	133 (503)	0	0	0
103-AP	DN	1,134 (4,292)	1,134 (4,292)	0	0	0
104-AP	DN	20 (76)	20 (76)	0	0	0
105-AP	DSSF	824 (3,119)	824 (3,119)	0	0	0
106-AP	DN	1,132 (4,285)	1,132 (4,285)	0	0	0
107-AP	DN	1,124 (4,254)	1,124 (4,254)	0	0	0
108-AP	DN	892 (3,376)	892 (3,376)	0	0	0
101-AW	DSSF	1,126 (4,262)	1,042 (3,944)	0	84 (318)	0
102-AW	DN	1,036 (3,921)	1,035 (3,917)	0	1 (4)	0
103-AW	DN/PD	649 (2,456)	286 (1,083)	0	363 (1,374)	0
104-AW	DN	1,125 (4,258)	835 (3,160)	0	179 (678)	111 (420)
105-AW	DN/PD	901 (3,410)	604 (2,286)	0	297 (1,124)	0
106-AW	DN	526 (1,991)	230 (871)	0	198 (749)	85 (322)
101-AY	DC	940 (3,558)	858 (3,248)	0	83 (314)	0
102-AY	DN	406 1,537)	374 (1,416)	0	32 (121)	0
101-AZ	Aging	947 (3,584)	912 (3,452)	0	35 (132)	0
102-AZ	Aging	969 (3,668)	878 (3,323)	0	91 (344)	0
101-SY	CC	1,107 (4,190)	17 (64)	530 (2,006)	0	560 (2,120)
102-SY	DN/PT	677 (2,562)	606 (2,294)	0	71 (269)	0
103-SY	CC	746 (2,824)	169 (640)	573 (2,169)	0	4 (15)

[a]See next page for description.
[b]Includes interstitial liquid.

Table 3-1. Double-Shell Tank Inventory as of February 1992. (2 sheets)

Waste type abbreviation	Waste type	Description
Aging	Aging waste	High-level, first cycle solvent extraction waste from PUREX (NCAW).
CC	Concentrated complexant	Concentrated produce from the evaporation of dilute complexed waste.
CP	Concentrated phosphate	Waste originating from the decontamination of 100 N Area Reactor. Concentration of this waste produces concentrated phosphate waste.
DC	Dilute complexed	Characterized by a high content of organic carbon including organic complexants: EDTA, citric acid, and HEDTA are the major complexants used. Main sources of DC waste are saltwell liquid inventory.
DN	Dilute noncomplexed	Low-activity liquid waste originating from T and S Plants, the 300 and 400 Areas, PUREX facility (decladding supernate, and miscellaneous wastes), 100 N Area (sulfate waste), B Plant, saltwells, and PFP (supernate).
DSS	Double-shell slurry	Waste evaporated almost to its sodium aluminate saturation boundary or 6.5 molar hydroxide in the evaporator. For reporting purposes, DSS is considered a solid.
DSSF	Double-shell slurry feed	Waste evaporated just before reaching the sodium aluminate saturation boundary of 6.5 molar hydroxide in the evaporator. This form is not as concentrated as double-shell slurry.
PN/PD	PUREX decladding	PUREX Neutralized Cladding Removal Waste (NCRW)is the solids portion of the PUREX Facility neutralized cladding removal waste stream, received in tank farms as a slurry. Classified as TRU waste.
PT	PFP TRU Solids	TRU solids from 200 West Area operations.

EDTA = ethylenediaminetetraacetic acid
HEDTA = hydroxyethylenediaminetriacetic acid
NCAW = neutralized cladding
PFP = Plutonium Finishing Plant
PUREX = Plutonium-Uranium Extraction (Plant)
TRU = transuranic (waste).

aluminate phase boundary. The hot slurry is pumped to a DST where it forms solids as it cools. The waste is then called DSS.

3.2.2 Planned Treatment of Double-Shell Slurry Feed and Double-Shell Slurry

The DSSF will be pumped from DSTs to the Grout Treatment Facility (GTF) for treatment and conversion into grout. The DSS will be treated in the same manner, except for one additional treatment step to remove the DSS solids from the DSTs.

Milestone M-01-01 of the Tri-Party Agreement (Ecology et al. 1990) calls for the completion of three grout campaigns of DST waste. One campaign of phosphate-sulfate LLW has been completed. The remaining two campaigns will use DSSF and DSS.

Grout treatment of DSSF and DSS will begin when the ongoing construction of vaults to contain these LLWs is completed.

Treatment of DSSF and DSS has been studied in the laboratory as part of the Grout Formulation Program to develop and qualify grout formulae for the solidification of the Hanford Site's DST waste. A formula consists of measured quantities of up to four dry materials (e.g., calcium carbonate, fly ash, blast-furnace slag, and cement), up to three liquid additives, and DSSF or DSS waste. The dry materials are blended together and then the liquids are added to the solids.

Qualification consists of verifying grout performance as a function of the following expected process variabilities:

- Changes in DSSF and DSS waste composition
- Dry material composition variables
- Changes in dry material storage conditions
- Dry material blending variables
- Variables in the mixing of DSSF and DSS waste with the dry blend
- Variables in grout curing conditions
- Changes in the long-term vault conditions (grout aging).

Grout formulation qualifications are expected to be completed in 1992.

3.3 PLANNED TREATMENT OF NEUTRALIZED CURRENT ACID WASTE

3.3.1 Definition of Neutralized Current Acid Waste

The NCAW is the aqueous high-salt waste from the first-cycle solvent extraction column in the PUREX Plant. This waste is neutralized to prevent corrosion of the tank farm carbon-steel tanks.

3.3.2 Planned Treatment Process of Neutralized Current Acid Waste

The first step in the proposed treatment process is to separate the solids from the supernatant (Figure 3-1) (Karnesky 1990a, 1990b). Solid-liquid separation has been demonstrated in the laboratory using a settle-decant process (Wong 1989). The solid-liquid separation step has previously been demonstrated in a plant test.

The supernatant contains most of the cesium that will be removed by ion exchange leaving a LLW fraction destined for the GTF. Cesium will be eluted from the ion-exchange column and combined with the solids from the initial solid-liquid separation step to form the HLW fraction of the NCAW destined for the HWVP.

3.3.3 Schedule

The NCAW treatment technology has been demonstrated in the laboratory. Plant-scale testing in Vault 244-AR and B Plant was scheduled to begin in October 1993. However, as a result of recent tank waste disposal program redefinition studies in 1991, it was recommended that B Plant, 244-AR Vault, and other existing Hanford processing facilities be excluded from further consideration as pretreatment processing facilities because of the high risk in achieving environmental and safety compliance (Grygiel et al. 1991). A revised schedule for pilot plant operations needed to support HWVP melter tests will be developed on the basis of an ongoing tank waste disposal program rebaselining activity to be completed in 1992. The development of a revised program baseline responds to the Secretary of Energy's Decision Statement dated December 28, 1991, to resolve an urgent program need to resolve Hanford tank waste safety issues and to prepare high-level radioactive defense waste for final treatment in grout and borosilicate glass form (DOE 1991).

3.4 PLANNED TREATMENT OF NEUTRALIZED CLADDING REMOVAL WASTE

3.4.1 Definition of Neutralized Cladding Removal Waste

Cladding removal waste (CRW) results from the dissolution of the N Reactor spent-fuel zircaloy cladding using the zirflex process in the PUREX Plant. Neutralization of this waste causes most of the zirconium to precipitate as a hydrated oxide, essentially removing all of the actinides and fission products from the solution. However, sufficient quantities of fine plutonium particles are entrained with the precipitated zirconium that the waste collected in the DSTs is considered to be a TRU waste. The waste sludge and supernate as stored in the DSTs is known as NCRW.

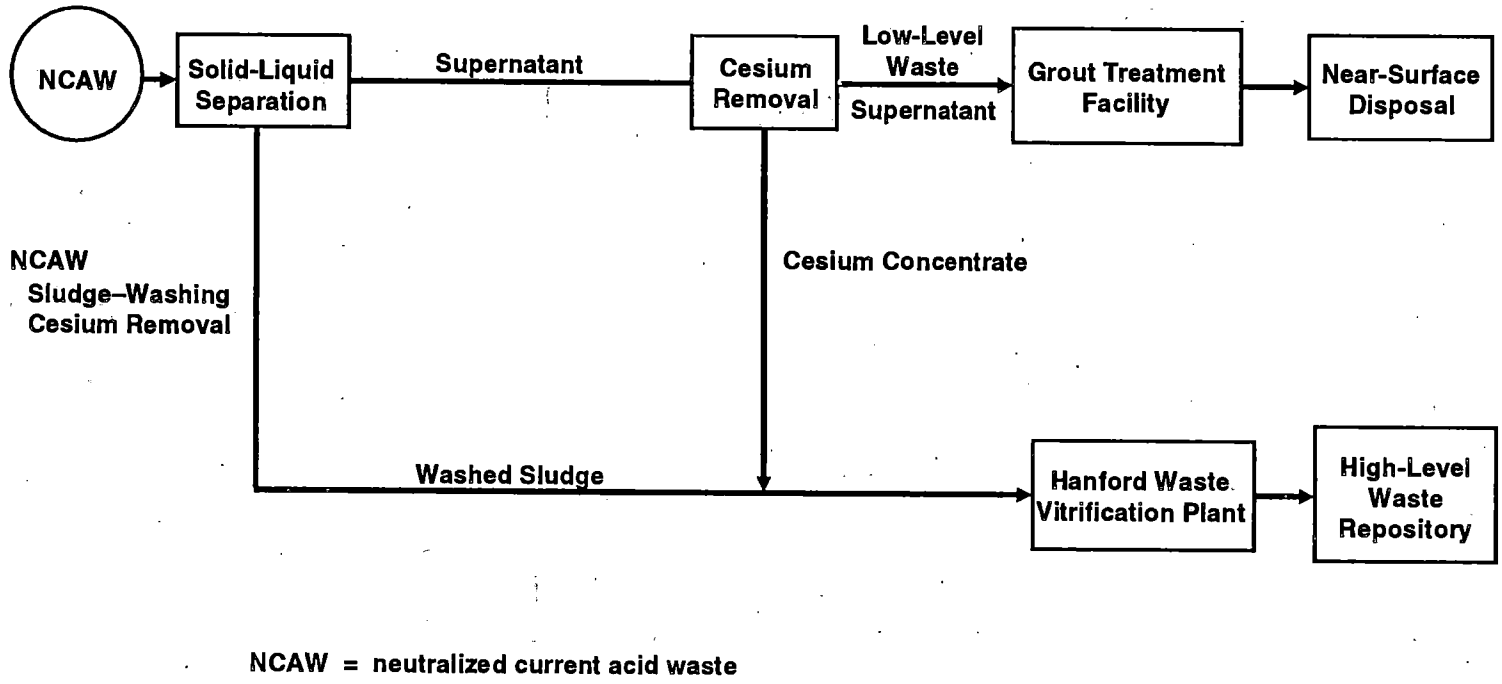

Figure 3-1. Neutralized Current Acid Waste Pretreatment and Disposal.

3.4.2 Planned Treatment Process of Neutralized Cladding Removal Waste

The first step in the proposed treatment process is to separate the solids from the supernate (Figure 3-2). The supernate is a LLW that can be sent to the GTF for further treatment (Kurath and Yeager 1987).

The solids from the liquid-solid separation step are then washed to remove soluble sodium and potassium compounds. The wash liquids are LLWs that can be sent to the GTF for further treatment. Although a processing step has not been selected to treat these solids, one promising approach consists of dissolving the solids with nitric acid and hydrofluoric acid. The dissolved TRU elements are then separated from the remaining undissolved solids and constitute the feed stream for the transuranium extraction (TRUEX) process.

The TRUEX process separates a small volume of the concentrated TRU waste from a large-volume LLW stream. The LLW stream is sent to the GTF. The concentrated TRU stream is recombined with the undissolved solids remaining from the previous acid dissolution step for transfer to the HWVP for vitrification.

3.4.3 Schedule

In FY 1991, pilot plant tests with NCRW were scheduled through FY 1996. Operation of the full-scale TRUEX process using a NCRW feed is currently being studied and a revised schedule will be issued in 1993 to reflect the results of the previously-cited program rebaselining effort.

3.5 PLANNED TREATMENT OF PLUTONIUM FINISHING PLANT WASTE

3.5.1 Definition of Plutonium Finishing Plant Waste

The PFP waste originates from the conversion of plutonium nitrate to oxide or metal and includes TRU laboratory wastes. The PFP waste also includes Plutonium Reclamation Facility (PRF) waste consisting of high-salt solvent extraction waste and organic wash waste.

3.5.2 Planned Treatment Process of Plutonium Finishing Plant Waste

The first step in the proposed treatment process is to separate the solids from the supernate (Figure 3-3). The supernate is a LLW that can be sent to the GTF for further treatment.

Although a treatment process has not been selected, one promising process is acid dissolution followed by treatment employing the TRUEX process. Another alternative would be to selectively leach critical components such as chromium from the sludge to minimize the number of glass canisters produced.

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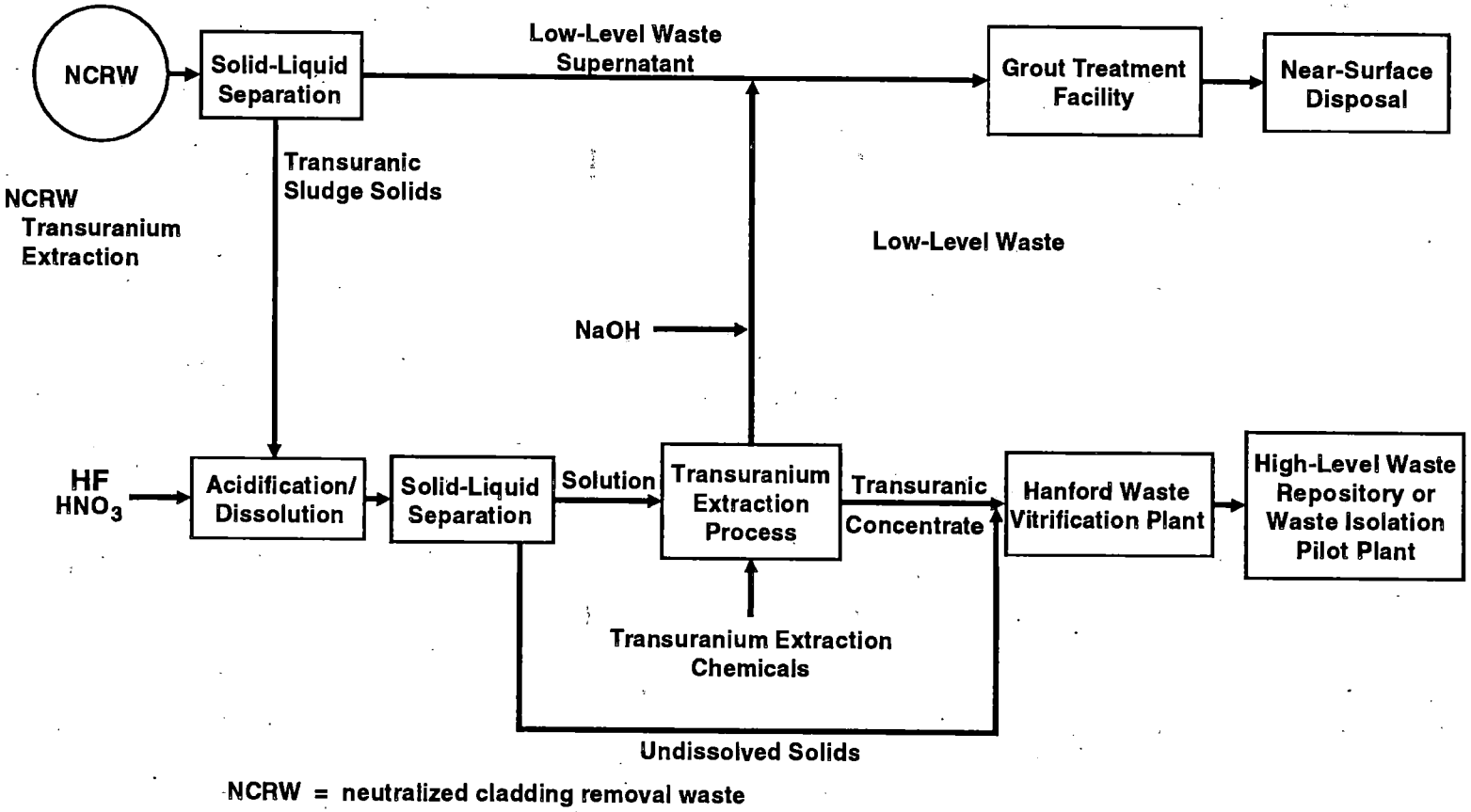

Figure 3-2. Neutralized Cladding Removal Waste Pretreatment and Disposal.

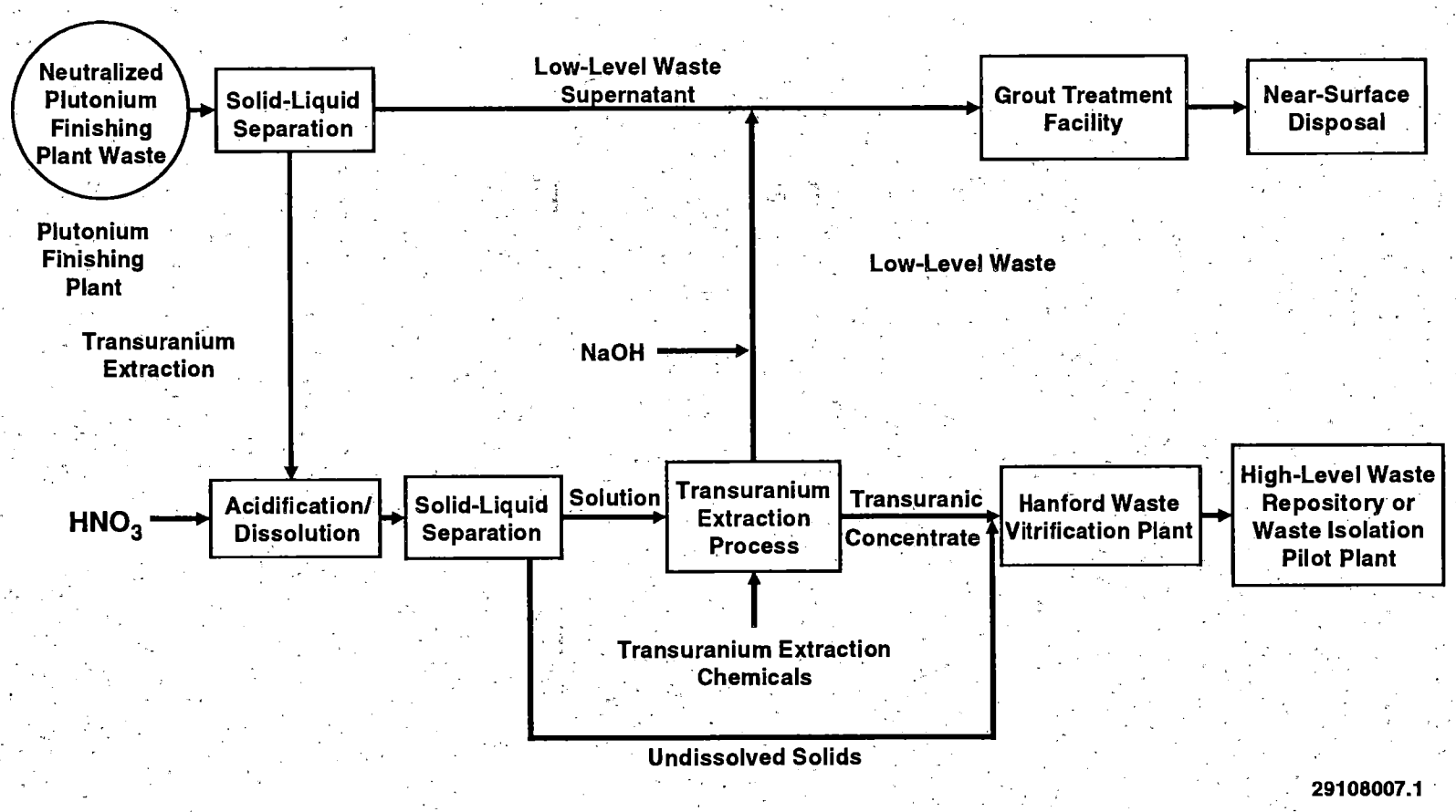

Figure 3-3. Plutonium Finishing Plant Waste Pretreatment and Disposal.

3.5.3 Schedule

In FY 1991, pilot plant testing of the PFP waste treatment flowsheet using the TRUEX process was scheduled for FY 1997. However, the current tank waste disposal rebaselining activity will develop updated schedules for the PFP waste treatment pilot plant testing in 1993.

3.6 PLANNED TREATMENT OF COMPLEXANT CONCENTRATE WASTE

3.6.1 Definition of Complexant Concentrate Waste

Complexant concentrate waste results from concentration of wastes containing large amounts of organic complexing agents. These organic compounds were introduced to the waste during strontium recovery processing in B Plant.

3.6.2 Planned Treatment Process of Complexant Concentrate Waste

During 1991, the goal of treatment was given the added scope to resolve the safety issues of watch list tanks by destroying organics and ferrocyanides. Two of the watch list tanks (101-SY and 103-SY) are complexed wastes in DSTs. Because the resolution of safety issues has priority over preparing grout and glass feeds, these tanks will be treated first by destroying the organics using one of several oxidation processes currently being evaluated. After removing cesium from the liquid phase of the oxidized waste, the remaining liquid is a candidate for grouting. The sludge may undergo further pretreatment. The extent of the pretreatment has not yet been determined. One possible treatment approach consists of acid dissolution followed by the TRUEX process. Other CC waste may not be oxidized initially.

Another possible process that has been investigated to some extent for CC waste is described as follows.

The first step in the proposed treatment process is to acidify the CC waste stream to dissolve as many of the solids as possible as shown in Figure 3-4 (Kurath 1985, 1986). The liquid is separated from the undissolved solid from the previous acid dissolution step and is then used as a feed stream to the TRUEX process. Complexant destruction may be performed before TRUEX processing, but is not required at this step in the treatment process at the present time.

The TRUEX process separates a low-volume TRU concentrate waste stream from a high-volume LLW stream containing organics and possibly cesium. The TRU concentrate stream is added to the remaining undissolved solids from the liquid-solid separation step following the initial acid dissolution step, as shown in Figure 3-4, and is then treated in the HWVP.

The LLW stream containing organics and cesium undergoes further treatment for organic destruction if not done previously. The LLW is then neutralized

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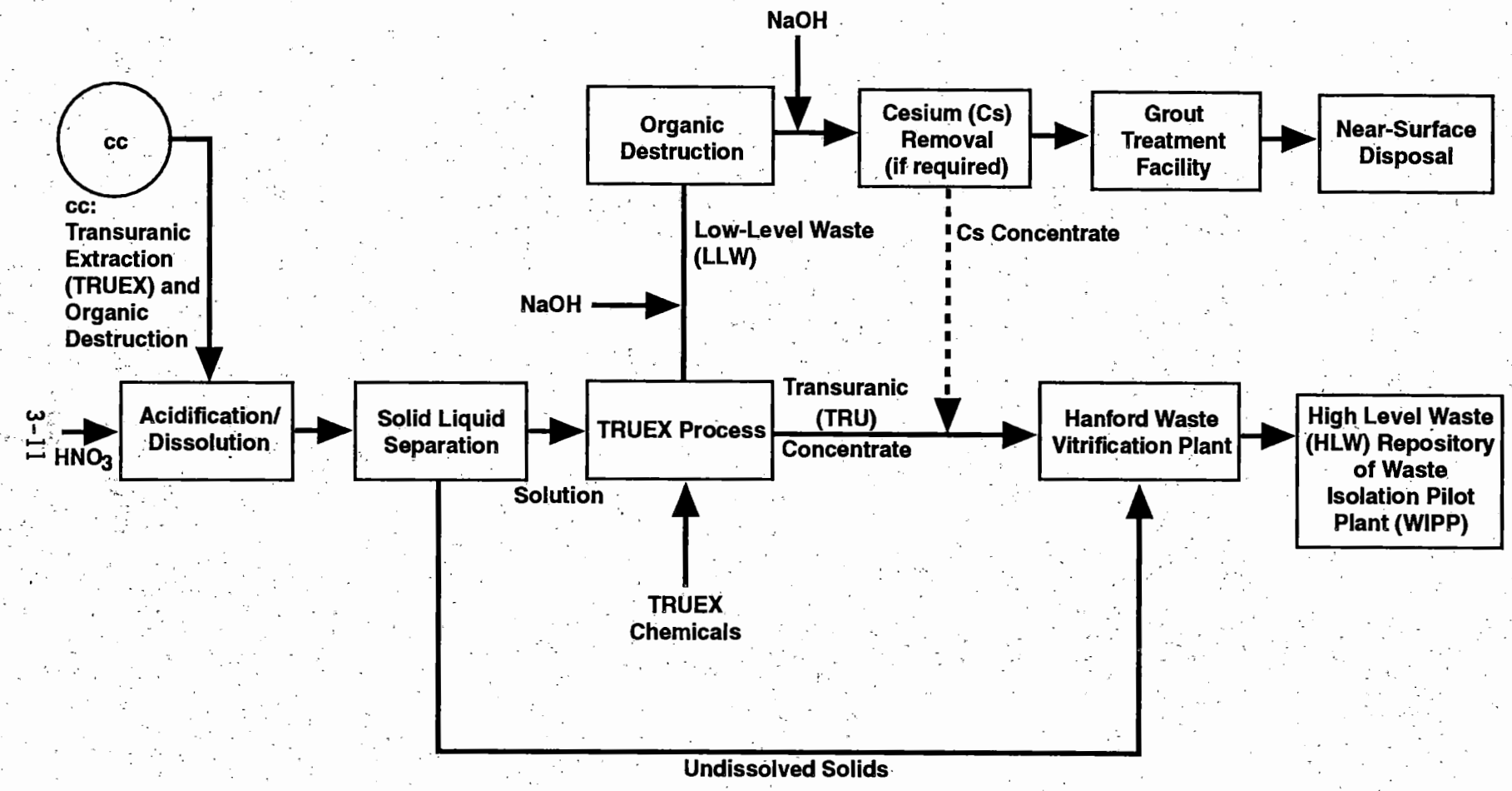

Figure 3-4. Complexant Concentrate Waste Pretreatment and Disposal.

and the cesium is removed (Lutton et al. 1980). The resulting LLW stream is sent to the GTF for conversion into grout. The cesium containing stream is sent to the HWVP.

Other alternatives to the TRUEX process also are being pursued. These include other solvent extraction processes, precipitation processes and the use of solid sorbents.

3.6.3 Schedule

In the FY 1991 tank waste treatability report (Giese 1991), pilot plant testing of the CC waste treatment process was scheduled for FY 1997 through FY 1999. However, a new schedule will be developed in 1992 to reflect the results of the ongoing rebaselining development. The full-scale processing schedule for CC waste also is currently being reviewed to evaluate the impact of cesium removal from the low activity portion of the treated waste on the overall treatment of CC wastes.

3.7 SUMMARY OF DOUBLE-SHELL TANK WASTE TREATMENT

Studies have been performed to evaluate alternative processes and facilities for treatment of DST wastes before final disposal. A 1989 study confirmed the technical and economic incentives for partitioning the waste into a large, low-level fraction suitable for near-surface disposal, and a smaller fraction of TRU waste and/or HLW that must be immobilized by solidification in glass (Kupfer et al. 1989).

An evaluation of alternative facilities for performing waste treatment processes and optimum schedules for timely completion of the DST waste disposal mission was completed in 1990. The evaluation defined the existing baseline waste treatment plan for DST waste at that time.

- Separate NCAW sludges from supernatant liquids and wash the sludge with water to remove soluble salts.
- Remove TRU waste components from acidified wastes using the TRUEX process. This technology is being pursed for application with NCRW, PFP waste, and CC waste as well as other alternatives.
- Remove cesium from alkaline NCAW and CC supernatant liquors.
- Destroy the complexant in CC waste to remove complexed TRU elements and provide a feed for near-surface disposal.

The ongoing experimental program (Swanson 1991a) provided process information in the areas of sludge retrieval, solvent extraction feed stability, dissolver residue compositions, and simulant properties. These areas of interest and the pertinent findings are summarized below.

- Investigations were performed to evaluate the amount of nitrogen oxides liberated in the NCRW pretreatment process. It is reported that an inhibitor will aid in affecting a rapid reaction of nitrogen

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oxides into less hazardous materials. Nitrogen compounds will be liberated in the dissolver only, rather than throughout the entire process, reducing offgas treatment problems.

- The composition of the dissolver residue, the primary feed to the HWVP, was characterized. This information will predict the HWVP glass composition.

- Because it is determined that total mixing of the NCRW waste tanks will not be feasible, work has been initiated to evaluate the processability of the various layers of sludge within the tanks. As a result, a problem has been identified with the feed stream to the solvent extraction section of the process. It has been found that these streams may form a solid precipitate under certain conditions, which would impact the effectiveness of the process. Several flow sheet variations were proposed to deal with the precipitation issue. This issue will be addressed further in subsequent studies.

- An evaluation of the stream that will be fed to the HWVP found that the NCRW pretreatment process added significant amounts of phosphate to this stream from the stripping agent used in the TRUEX process. As a result, alternate stripping agents for TRUEX process are considered. The results of these tests suggest that the phosphate can be reduced significantly by using sodium carbonate as an additional additive in the stripping agent.

A design base experiment was performed (Swanson 1991b) which confirmed the applicability of the dissolution/TRUEX process for pretreating NCRW. The design base experiment was essentially based on the current flowsheet. It did not include washing of the NCRW sludge. The experiment demonstrated that about 95 percent of the waste materials end up as LLW, while more than 99 percent of the TRUs end up in the HWVP feed.

Recent accomplishments include:

- Completion of the conceptual design report for the pilot-scale facility for demonstrating the TRUEX process with actual DST wastes (KEH 1991).

- Ozone-ultraviolet light methods for organic complexant destruction were found to be less effective at complexant destruction than the use of hydrogen peroxide.

Additional waste treatability tasks that are in progress or expected to be initiated in FY 1993 are described below. Documentation describing the results of these studies will be provided in future annual reports.

- Continue laboratory-scale tests to assess the application of the TRUEX process to remove TRU components from acidic solutions of actual NCRW, PFP waste, and CC waste.

- Proceed with the identification of the TRUEX pilot plant needs.

- Continue laboratory-scale tests of complexant destruction methods. Efforts will focus on wet oxidation, further use of ozone as an oxidant, and calcination.

- Provide updated preliminary conceptual flowsheets for the TRUEX process for pretreatment of NCRW, PFP waste, and CC waste.

- Perform capacity tests of candidate ion-exchange resins for removal of ^{137}Cs from alkaline waste.

3.8 TREATMENT OF WASTE AFTER PRETREATMENT ACTIVITIES

3.8.1 Grout Treatment

Grout treatment is the process of mixing selected DST wastes with grout-forming solids, and possibly with liquid chemical additives, to form a grout slurry that is pumped into near-surface lined concrete vaults for solidification and permanent disposal. The waste is characteristically corrosive because of the high hydroxide ion concentration and is characterized as toxic because of the high concentrations of nitrite and hydroxide ions.

The grout disposal vaults are considered disposal facilities and are treated as surface impoundments until final closure as landfills.

3.8.2 Hanford Waste Vitrification Plant Project

3.8.2.1 The Hanford Waste Vitrification Plant. The HWVP will immobilize high-level Hanford Site defense wastes by vitrification. In the slurry receipt and adjustment tank (SRAT), dilute pre-treated feed will be concentrated into a slurry by evaporation and chemically adjusted to facilitate slurry transport. In the slurry mix evaporator (SME) tank, glass formers will be added in the form of a frit to the slurry, and the slurry will be further concentrated and chemically adjusted before being transferred to the melter feed tank (MFT). The MFT feed will be fed to a joule-heated glass melter. The molten glass product will be poured into stainless steel canisters that will be sealed, decontaminated, and then stored until future shipment to a permanent waste repository. Figure 3-5 provides a process flow schematic diagram for the HWVP.

Single-shell tank waste is to be considered for vitrification. The glass formulations and plant design for the current baseline program are based on the processing of HLW from the DSTs. The DST wastes to be vitrified include NCAW, NCRW, CC, and PFP waste. Adequate design flexibility is being incorporated to facilitate future waste immobilization objectives. The feasibility of, and requirements for vitrifying other high-level Hanford Site defense wastes are under study and are discussed in Section 6.0.

The HWVP process and storage facilities are designed for a 40-yr operating lifetime and also are being designed to remain functional after a design basis accident caused by certain natural phenomena; i.e., seismic disturbances (earthquakes), tornadoes, or ash fall from volcanic eruptions.

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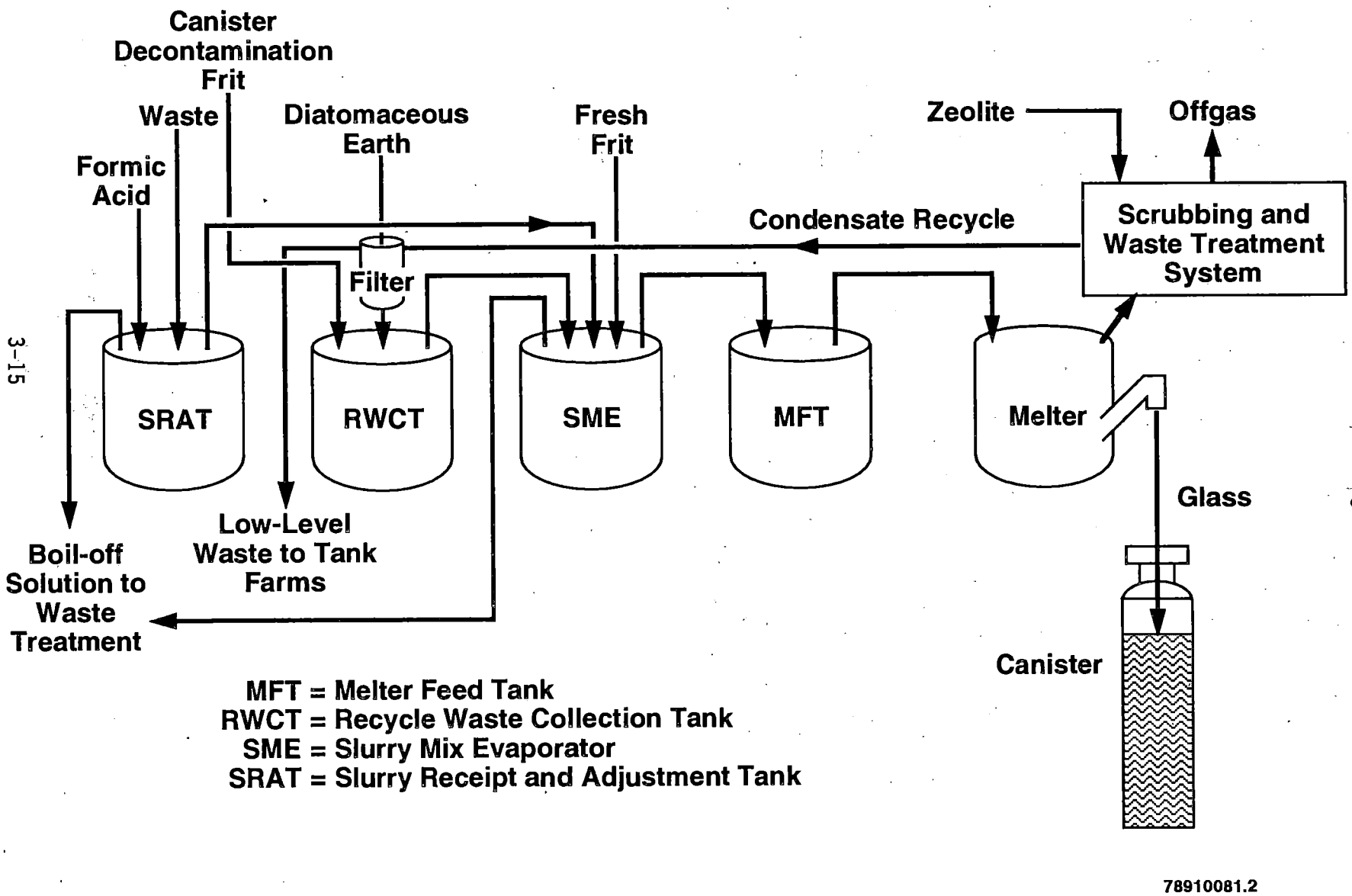
Figure 3-5. Hanford Waste Vitrification Plant Glass Production-Process Flow Schematic Diagram.



The facilities provide for remote operation and maintenance of the process with appropriate biological shielding for operator safety. Heating, ventilation, and air conditioning systems provide additional confinement barriers to limit any potential spread of radioactive contaminants.

The vitrification process is comprised of five major subsystems which will include the feed receipt and preparation system, melter system, offgas treatment system, canister closure and decontamination system, and the waste handling system. The canister storage system, which was formerly a proposed subsystem, will be a separate facility relative to the HWVP project. The vitrification process subsystems will be remotely operated and maintained and will be located within process cells in the vitrification building. Cold chemical storage, utility systems, and personnel support services required to support the vitrification process will be located within buildings adjacent to the vitrification building. Wastes from the process and process support operations will be treated within the HWVP and non-TRU wastes will be discharged outside of the HWVP to the underground waste holding tank. The current baseline for HWVP startup date is December 1999, with cold operational testing and qualification testing scheduled during the preceding 18-month period.

3.8.2.2 Waste Feed Processability. The HWVP will process a number of different feed types, whose composition may not be fully characterized prior to the initial hot start up of the plant in December 1999. A composition variability study (CVS) is being conducted to characterize the relationship between glass composition and glass properties. The ability of the HWVP to produce a molten glass acceptable to melter operation and a glass product acceptable to the permanent geologic repository is controlled by a number of properties and features including viscosity, electrical resistivity, thermal expansion, crystallinity, durability, liquidus temperature, radioactivity, heat generation, and concentration of key components that may limit waste oxide loading in the glass. The current strategy, which provides maximum flexibility for handling variations in composition, is to define an envelope of glass compositions. This approach will be used to help determine the optimum waste oxide loading for all the vitrified waste forms; i.e., DST wastes (e.g., PFP, NCRW, CC, NCAW) and applicable SST wastes.

4.0 TREATMENT OF EXISTING SINGLE-SHELL WASTES

4.1 DESCRIPTION OF SINGLE-SHELL TANK WASTES

One hundred and forty-nine SSTs contain portions of HLW, TRU waste, and LLW produced during Hanford Site operations before 1980. The current waste inventory of the SST system as of February 1992 is given in Table 4-1, which is taken from the *Tank Farm Surveillance and Waste Status Summary Report for February 1992* (Hanlon 1992). Interim stabilization efforts are currently underway to remove pumpable liquid from the SSTs leaving saltcake, sludge, and interstitial liquid. This supports Tri-Party Agreement interim milestone M-05-09 (Ecology et al. 1990). The remaining SST contents form the basis for future treatment efforts.

4.2 TREATMENT OF SINGLE-SHELL TANK WASTES

The major SST treatment objectives are to resolve the tank safety issues pertaining to hydrogen generation, organic compounds, and ferrocyanide compounds, which can potentially react to evolve both heat and toxic gases (Borsheim and Kirch 1991). Two treatment alternatives are being considered; in situ treatment and treatment after retrieval.

The treatment-after-retrieval alternative has two additional goals; (1) minimize the volume of waste feed to the HWVP while meeting current DST feed chemical concentration limits, and (2) maximize the fraction of nonradioactive chemical compounds routed to GTF while meeting the non-TRU (<100 nCi/g), ^{90}Sr, and ^{137}Cs, repository concentration requirements for the solidified grout (Boomer 1991). The processes for treatment of the retrieved SST waste are currently based on the processes and equipment being developed for the DST program; e.g., sludge washing, TRUEX, cesium ion-exchange, and possibly complexant destruction. Treatment technologies specific to SST waste are being studied and funded by the DOE Environmental Restoration (EM-40) Program and the Office of Technology Development (OTD) (EM-50) Program, such as the Underground Storage Tank/Integrated Demonstration (UST/ID).

One additional tank safety issue pertains to a single SST (tank 106-C), which evolves sufficient radioactive decay heat to require periodic additions of cooling water. Currently, a total of 51 SSTs have Priority I related safety issues; i.e., 18 tanks with potential for hydrogen or flammable gas accumulation above the flammability limit, 24 tanks containing more than 1,000 g-mol of ferrocyanide, 8 tanks with high organic content, and the aforementioned single high-heat tank (Wilson and Reep 1991).

4.3 STATUS OF SINGLE-SHELL TANK WASTE TREATMENT STUDIES

The following information provides the status of SST waste treatment activities completed or in progress. In many cases, activities being performed by the DST program also apply to the SST program.

Table 4-1. Single-Shell Tank Inventory as of February 1992. (6 sheets)

Tank Number	Waste material[a]	Volume in kgal (m^3)			
		Total waste	Supernatant	Sludge[b]	Salt cake
101-A	DSSF	953 (3,607)	0	3 (11)	950 (3,596)
102-A	DSSF	41 (155)	4 (15)	15 (57)	22 (83)
103-A	DSSF	370 (1,400)	4 (15)	366 (1,385)	0
104-A	NCPLX	28 (106)	0	28 (106)	0
105-A	NCPLX	19 (72)	0	19 (72)	0
106-A	CP	125 (473)	0	125 (473)	0
101-AX	DSSF	748 (2,831)	0	3 (11)	745 (2,820)
102-AX	CC	39 (148)	3 (11)	7 (26)	29 (110)
103-AX	CC	112 (424)	0	2 (8)	110 (416)
104-AX	NCPLX	7 (26)	0	7 (26)	0
101-B	NCPLX	113 (428)	0	113 (428)	0
102-B	NCPLX	32 (121)	4 (15)	18 (68)	10 (38)
103-B	NCPLX	59 (223)	0	59 (223)	0
104-B	NCPLX	371 (1,404)	1 (4)	301 (1,139)	69 (261)
105-B	NCPLX	306 (1,158)	0	40 (151)	266 (1,007)
106-B	NCPLX	117 (443)	1 (4)	116 (439)	0
107-B	NCPLX	165 (625)	1 (4)	164 (621)	0
108-B	NCPLX	94 (356)	0	94 (356)	0
109-B	NCPLX	127 (481)	0	127 (481)	0
110-B	NCPLX	246 (931)	1 (4)	245 (927)	0
111-B	NCPLX	237 (897)	1 (4)	236 (893)	0
112-B	NCPLX	33 (125)	3 (11)	30 (114)	0
201-B	NCPLX	29 (110	1 (4)	28 (106)	0
202-B	NCPLX	27 (102)	0	27 (102)	0
203-B	NCPLX	51 (193)	1 (4)	50 (189)	0
204-B	NCPLX	50 (189)	1 (4)	49 (185)	0
101-BX	NCPLX	43 (163)	1 (4)	42 (159)	0
102-BX	NCPLX	96 (363)	0	96 (363)	0
103-BX	NCPLX	66 (250)	4 (15)	62 (235)	0
104-BX	NCPLX	99 (375)	3 (11)	96 (364)	0
105-BX	NCPLX	51 (193)	5 (19)	43 (163)	3 (11)
106-BX	NCPLX	46 (174)	15 (57)	31 (117)	0
107-BX	NCPLX	345 (1,306)	1 (4)	344 (1,302)	0

Table 4-1. Single-Shell Tank Inventory as of February 1992. (6 sheets)

Tank Number	Waste material[a]	Volume in kgal (m^3)			
		Total waste	Supernatant	Sludge[b]	Salt cake
108-BX	NCPLX	26 (98)	0	26 (98)	0
109-BX	NCPLX	193 (731)	0	193 (731)	0
110-BX	NCPLX	199 (753)	1 (4)	189 (715)	9 (34)
111-BX	NCPLX	230 (870)	19 (72)	68 (257)	143 (541)
112-BX	NCPLX	165 (625)	1 (4)	164 (621)	0
101-BY	NCPLX	387 (1,465)	0	109 (413)	278 (1,052)
102-BY	NCPLX	341 (1,291)	0	0	341 (1,291)
103-BY	NCPLX	400 (1,514)	0	5 (19)	395 (1,495)
104-BY	NCPLX	406 (1,536)	0	40 (151)	366 (1,385)
105-BY	NCPLX	503 (1,904)	0	44 (167)	459 (1,737)
106-BY	NCPLX	642 (2,430)	0	95 (360)	547 (2,070)
107-BY	NCPLX	266 (1,007)	0	60 (227)	206 (780)
108-BY	NCPLX	228 (863)	0	154 (583)	74 (280)
109-BY	NCPLX	398 (1,506)	0	103 (390)	295 (1,116)
110-BY	NCPLX	398 (1,506)	0	103 (390)	295 (1,116)
111-BY	NCPLX	459 (1,737)	0	21 (79)	438 (1,658)
112-BY	NCPLX	291 (1,101)	0	5 (19)	286 (1,082)
101-C	NCPLX	88 (333)	0	88 (333)	0
102-C	DC	423 (1,601)	0	423 (1,601)	0
103-C	NCPLX	195 (738)	133 (503)	62 (235)	0
104-C	CC	295 (1,117)	0	295 (1,117)	0
105-C	NCPLX	150 (568)	0	150 (568)	0
106-C	NCPLX	229 (867)	32 (121)	197 (746)	0
107-C	DC	275 (1,041)	0	275 (1,041)	0
108-C	NCPLX	66 (250)	0	66 (250)	0
109-C	NCPLX	66 (250)	4 (15)	62 (235)	0
110-C	DC	187 (708)	0	187 (708)	0
111-C	NCPLX	57 (216)	0	57 (216)	0
112-C	NCPLX	104 (394)	0	104 (394)	0
201-C	NCPLX	2 (8)	0	2 (8)	0
202-C	EMPTY	1 (4)	0	1 (4)	0
203-C	NCPLX	5 (19)	0	5 (19)	0
204-C	NCPLX	3 (11)	0	3 (11)	0
101-S	NCPLX	427 (1,616)	12 (45)	244 (924)	171 (647)
102-S	DSSF	549 (2,078)	0	4 (15)	545 (2,063)

Table 4-1. Single-Shell Tank Inventory as of February 1992. (6 sheets)

Tank Number	Waste material[a]	Volume in kgal (m^3)			
		Total waste	Supernatant	Sludge[b]	Salt cake
103-S	DSSF	248 (939)	17 (64)	10 (38)	221 (837)
104-S	NCPLX	294 (1,113)	1 (4)	293 (1,109)	0
105-S	NCPLX	456 (1,726)	0	2 (8)	454 (1,718)
106-S	NCPLX	543 (2,055)	0	32 (121)	511 (1,934)
107-S	NCPLX	368 (1,393)	6 (23)	293 (1,109)	69 (261)
108-S	NCPLX	604 (2,286)	0	4 (15)	600 (2,271)
109-S	NCPLX	568 (2,150)	0	13 (49)	555 (2,101)
110-S	NCPLX	692 (2,619)	0	131 (496)	561 (2,123)
111-S	NCPLX	596 (2,256)	10 (38)	139 (526)	447 (1,692)
112-S	NCPLX	637 (2,411)	0	6 (23)	631 (2,388)
101-SX	DC	456 (1,726)	1 (4)	112 (424)	343 (1,298)
102-SX	DSSF	543 (2,055)	0	117 (443)	426 (1,612)
103-SX	NCPLX	652 (2,468)	1 (4)	115 (435)	536 (2,029)
104-SX	DSSF	614 (2,324)	0	136 (515)	478 (1,809)
105-SX	DSSF	683 (2,585)	0	73 (276)	610 (2,309)
106-SX	NCPLX	538 (2,036)	61 (231)	12 (45)	465 (1,760)
107-SX	NCPLX	104 (394)	0	104 (394)	0
108-SX	NCPLX	115 (435)	0	115 (435)	0
109-SX	NCPLX	250 (946)	0	250 (946)	0
110-SX	NCPLX	62 (235)	0	62 (235)	0
111-SX	NCPLX	125 (473)	0	125 (473)	0
112-SX	NCPLX	92 (348)	0	92 (348)	0
113-SX	NCPLX	26 (98)	0	26 (98)	0
114-SX	NCPLX	181 (685)	0	181 (685)	0
115-SX	NCPLX	12 (45)	0	12 (45)	0
101-T	NCPLX	133 (503)	30	103 (503)	0
102-T	NCPLX	32 (121)	13 (49)	19 (72)	0
103-T	NCPLX	27 (102)	4 (15)	23 (87)	0
104-T	NCPLX	445 (1,684)	3 (11)	442 (1,673)	0
105-T	NCPLX	98 (371)	0	98 (371)	0
106-T	NCPLX	21 (79)	2 (7)	19 (72)	0
107-T	NCPLX	180 (681)	9 (34)	171 (647)	0
108-T	NCPLX	44 (167)	0	44 (167)	0
109-T	NCPLX	58 (220)	0	58 (220)	0
110-T	NCPLX	379 (1,435)	3 (12)	376 (1,423)	0

Table 4-1. Single-Shell Tank Inventory as of February 1992. (6 sheets)

Tank Number	Waste material[a]	Volume in kgal (m^3)			
		Total waste	Supernatant	Sludge[b]	Salt cake
111-T	NCPLX	458 (1,734)	2 (7)	456 (1,727)	0
112-T	NCPLX	67 (253)	7 (26)	60 (227)	0
201-T	NCPLX	29 (110)	1 (4)	28 (106)	0
202-T	NCPLX	21 (79)	0	21 (79)	0
203-T	NCPLX	35 (132)	0	35 (132)	0
204-T	NCPLX	38 (144)	0	38 (144)	0
101-TX	NCPLX	87 (329)	3 (11)	84 (318)	0
102-TX	NCPLX	113 (428)	0	0	113 (428)
103-TX	NCPLX	157 (594)	0	157 (594)	0
104-TX	NCPLX	65 (246)	1 (4)	0	64 (242)
105-TX	NCPLX	609 (2,305)	0	0	609 (2,305)
106-TX	NCPLX	453 (1,715)	0	0	453 (1,715)
107-TX	NCPLX	36 (136)	1 (4)	0	35 (132)
108-TX	NCPLX	134 (507)	0	0	134 (507)
109-TX	NCPLX	384 (1,453)	0	0	384 (1,453)
110-TX	NCPLX	462 (1,749)	0	0	462 (1,749)
111-TX	NCPLX	370 (1,400)	0	0	370 (1,400)
112-TX	NCPLX	649 (2,456)	0	0	649 (2,456)
113-TX	NCPLX	607 (2,297)	0	0	607 (2,297)
114-TX	NCPLX	535 (2,025)	0	0	535 (2,025)
115-TX	NCPLX	640 (2,422)	0	0	640 (2,422)
116-TX	NCPLX	631 (2,388)	0	0	631 (2,388)
117-TX	NCPLX	626 (2,369)	0	0	626 (2,369)
118-TX	NCPLX	347 (1,313)	0	0	347 (1,313)
101-TY	NCPLX	118 (447)	0	118 (447)	0
102-TY	NCPLX	64 (242)	0	0	64 (242)
103-TY	NCPLX	162 (613)	0	162 (613)	0
104-TY	NCPLX	46 (174)	3 (11)	43 (163)	0
105-TY	NCPLX	231 (874)	0	231 (874)	0
106-TY	NCPLX	17 (64)	0	17 (64)	0
101-U	NCPLX	25 (95)	3 (11)	22 (84)	0
102-U	NCPLX	374 (1,416)	18 (68)	43 (163)	313 (1,185)
103-U	NCPLX	468 (1,771)	13 (49)	32 (121)	423 (1,601)
104-U	NCPLX	122 (462)	0	122 (462)	0
105-U	NCPLX	418 (1,582)	37 (140)	32 (121)	349 (1,321)

Table 4-1. Single-Shell Tank Inventory as of February 1992. (6 sheets)

Tank Number	Waste material[a]	Volume in kgal (m^3)			
		Total waste	Supernatant	Sludge[b]	Salt cake
106-U	NCPLX	226 (855)	15 (57)	26 (98)	185 (700)
107-U	DSSF	406 (1,537)	31 (117)	15 (57)	360 (1,363)
108-U	NCPLX	468 (1,771)	24 (90)	29 (110)	415 (1,571)
109-U	NCPLX	463 (1,753)	19 (72)	48 (182)	396 (1,499)
110-U	NCPLX	186 (704)	0	186 (704)	0
111-U	DSSF	329 (1,245)	0	26 (98)	303 (1,147)
112-U	NCPLX	49 (185)	4 (15)	45 (170)	0
201-U	NCPLX	5 (19)	1 (4)	4 (15)	0
202-U	NCPLX	5 (19)	1 (4)	4 (15)	0
203-U	NCPLX	3 (11)	1 (4)	2 (7)	0
204-U	NCPLX	3 (11)	1 (4)	2 (7)	0

[a]See next page for description.
[b]Includes interstitial liquid.

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Table 4-1. Single-Shell Tank Inventory as of February 1992. (6 sheets)

Waste type abbreviation	Waste type	Description
CC	Concentrated complexant	Concentrated product from the evaporation of dilute complexed waste.
CP	Concentrated phosphate	Waste originating from the decontamination of 100 N Reactor. Concentration of this waste produces concentrated phosphate waste.
DC	Dilute complexed	Characterized by a high content of organic carbon including organic complexants. EDTA, citric acid, HEDTA, and IDA are the major complexants used. Main sources of DC waste are saltwell liquid inventory.
DSSF	Double-shell slurry feed	Waste evaporated just before reaching the sodium aluminate saturation boundary of 6.5 molar hydroxide in the evaporator. This form is not as concentrated as double-shell slurry.
NCPLX	Noncomplexed	General waste term applied to all Hanford Site liquors not identified as complexed.

EDTA = ethylenediaminetetraacetic acid
HEDTA = hydroxyethylenediaminetriacetic acid
IDA = iminodiacetate

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4.3.1 Removal of Organic and Ferrocyanide Components

Several promising processes are currently under evaluation and/or testing for the removal of organic and ferrocyanide compounds from Hanford Site tank wastes. One of these processes involves oxidizing the organic waste with ozone at ambient conditions of temperature and pressure to destroy the organic constituents of the waste. Ozonation is a process that could possibly degrade organic and ferrocyanide compounds sufficiently to resolve safety concerns and does not add to the current volume of waste or require chemical additions other than the ozone oxidizer.

A laboratory-scale ozone reactor is being used to demonstrate the destruction of organic compounds and ferrocyanide compounds contained in Hanford tank waste. Preliminary results indicate that the reactor can successfully destroy the compounds affecting tank safety. Experiments with simulated tank waste indicate, however, that a significant amount of ozone is required to degrade nickel ferrocyanide, the form found in Hanford tank wastes, than is required for organic compounds. If validated by future testing, this could make the process economically unattractive for ferrocyanide destruction.

Calcination is a processing alternative that is also being considered for this application. In this process, the waste is heated to dryness, and then to temperatures sufficient to oxidize organic and ferrocyanide compounds in the waste. Calcination processes are used in a variety of applications at temperatures varying from about 300 °C to 1,700 °C. The process typically produces a solid oxide product and offgases both inorganic and organic volatile combustion products of lower molecular weight. Thus, calcination can possibly reduce the volume of radioactive waste requiring disposal. However, this process may be difficult to apply to the high sodium-containing Hanford Site tank wastes. When wastes containing high concentrations of sodium are calcined, the sodium melts and agglomerates into a product that is difficult to process.

A calcination/dissolution process has been demonstrated that resolves tank safety issues and separates the TRUs into a relatively small volume. The results from testing small quantities of actual radioactive tank waste indicate that a calcination/dissolution process is feasible. A full-scale demonstration is planned for later in 1992 or early 1993 that will calcine 2,270 kg (5,000 lb) of simulated waste to determine the feasibility of scaleup.

Other organic destruction concepts being tested include: (1) ultrasonic wave (sonochemical) pyrolysis, (2) electrochemical oxidation, and (3) high pressure/temperature oxidation.

Ultrasonic wave (sonochemical) pyrolysis involves the excitation of an aqueous waste solution that generates micron-size cavitation bubbles that develop high temperatures and pressures [approximately 5000 °C and 490 kg/cm^2 (7,000 lb/in^2)] when they collapse. These conditions, while extremely short in duration, are known to produce several reactive species, including hydrogen peroxide and hydroxyl radical. These species, in turn, can degrade some organic compounds. The process operates at ambient temperature and pressure, requires minimal chemical additions (except to adjust the pH of the treated

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solution) and produces no secondary waste products (except for the offgases resulting from the oxidation of the organics). Sonochemical destruction of chlorinated hydrocarbons has been demonstrated in proof-of-principle tests in dilute solutions. Laboratory studies currently are planned at the University of Akron, Ohio, to evaluate the performance of this process with solutions of concentrated Hanford Site waste simulants.

In the electrochemical oxidation process, organic waste is introduced into an electrochemical cell containing high concentrations of nitric acid. The solution also contains a small quantity of silver, cerium, or other metal ion that in its higher oxidation state, is a kinetically strong, rapid oxidizing agent. The metal ions are oxidized at the cathode surface of the cell and then reduced by reacting with and oxidizing other materials, such as organic or ferrocyanide molecules. Unless this process can be modified for use in high pH (basic) solutions, it will suffer the disadvantages of increases in waste volume that are associated with acidification and subsequent reneutralization of the waste.

The supercritical water oxidation process involves pressurization and heating the waste solution above the critical point of the mixture. Above the critical point, the nitrate/nitrite present in the waste will oxidize the organics and ferrocyanides present. Rapid, high-efficiency waste oxidation reactions occur in the temperature range of about 400 to 600 °C and approximately 210 to 350 kg/cm^2 (3,000 to 5,000 lb/in^2). This process also has the potential to destroy nitrates and nitrites in the waste. Salts and metals precipitate out of the supercritical solution and can be subsequently treated.

4.3.2 Removal of Transuranic Components

The technology that was developed to remove the TRU waste content of the DST wastes, which was discussed in the previous section have direct application to the treatment of SST waste.

During this reporting period, americium, plutonium, and uranium ions were successfully removed from acidic tank waste solutions using several types of extractive chromatographic resins (Barney and Cowan 1992). Reagents also were tested for the dissolution of tank sludges to specifically accommodate subsequent TRU extraction (Schulz and Kupfer 1991).

4.3.3 Removal of Strontium and Cesium Components

In addition to the previously discussed technology for removal of strontium and cesium in DSTs (e.g., SREX, ion-exchange, etc.), testing has been completed on a novel separation technology known as Superlig*. Superlig is reportedly known for its ability to efficiently and selectively separate certain metal ions, including strontium and cesium. This technology utilizes

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*Superlig is a registered trademark of IBC Advanced Technologies, Inc., Provo, Utah.

macrocyclic ligands to selectively capture specific anions and has been used to remove trace metals from industrial waste waters (Camaioni et al. 1992).

4.3.4 General Pretreatment Testing

The following testing of several simulated tank waste recipes was completed by Westinghouse Hanford and Pacific Northwest Laboratory (PNL) (Jones et al. 1991) (Bloom et al. 1992).

- Sequential leaching of TRUs
- Separation of cesium by means of freeze crystallization
- Thermochemical reduction of nitrate ion.

4.4 ENGINEERING STUDIES

4.4.1 Initial Pretreatment Module

The Initial Pretreatment Module (IPM), Project W-236B, is being developed to comply with the direction and guidance contained in the Secretary, DOE, Decision Statement, dated December 20, 1991 (DOE 1991). The major objective of Project W-236B is to process Hanford Site tank wastes in such a manner as to resolve all watch list tank safety issues either by destroying or modifying the constituents (organics and ferrocyanides) that cause the safety concerns. A second objective of the project also addresses the removal of cesium to prepare waste for grout disposal thereby alleviating the tank space availability issues. Cesium removal also produces a vitrification process feed stream. The third objective of the facility is to provide a pilot plant capability to support the Tank Waste Remediation System (TWRS) program. A broad range of processing alternatives and facility options are being considered.

4.4.2 Comprehensive Treatment Studies

The 60 percent completion level of the systems engineering study for the closure of SSTs, issued in 1991, is continuing irrespective of evolving treatment priorities (Boomer et al. 1991). A program also has been initiated to evaluate the various alternatives for disposal of tank waste whereby performance will be measured using numerical models (Sonnichsen 1991). An earlier study was completed that documents the overall technology requirements, resources, equipment, program funding, and plans for closure of the SSTs (Klem 1990). Finally, the tank waste program redefinition which includes a systematic evaluation of the present status of the technical circumstances, alternatives, and regulatory issues for SSTs was completed during this reporting period (Grygiel et al. 1991).

4.4.3 Ferrocyanide Tank Studies

Several studies have been completed during this reporting period that relate to SSTs containing ferrocyanide (Cash and Dukelow 1992). The current plan is to further characterize the tanks containing ferrocyanide before making a treatment decision.

4.4.4 High-Heat Generating Tank Studies

Derived heat transfer relationships from the studies of those tanks with high-heat generation rates indicate that radiative heat transport through the air space in the tanks is higher than the heat transport via natural convection (Barker 1991a, 1991b).

4.4.5 In Situ Treatment Studies

Regulatory issues, technology development, and costs for in situ vitrification of tank wastes currently are being addressed in more detail (Corathers 1992) (Tixler et al. 1992). A baseline for dome fill technology, including an evaluation of potential fill materials, has been established (Smyth et al. 1992).

4.4.6 Characterization

A historical baseline for waste characterization of the SSTs has been completed (Droppo 1991). Recommendations for the design of a waste characterization program using a systems analysis technique have been developed (Buck et al. 1991). In addition, a sampling and laboratory analysis plan for the next ten tanks scheduled for this activity has been completed (Hill et al. 1991).

4.4.7 Grout Pretreatment Studies

Studies are underway to evaluate the need to remove radionuclides from tank waste before shallow land disposal (Worthington 1991). The study concludes that if grout can meet the existing regulatory requirements, no removal of contaminants is considered necessary for those wastes currently planned to be disposed of prior to the year 2001.

4.4.8 Tank Waste Retrieval

Technologies for retrieving wastes from SSTs have been identified for testing (Krieg et al. 1990). This study reviews current waste retrieval technologies including pumping, sluicing, air transport, and mechanical mixing.

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5.0 EVALUATION AND SELECTION OF GROUT

Cement-based grouts are extensively used in the United Sates (U.S.) and worldwide as a vehicle for immobilization and near-surface disposal of solid and liquid LLWs. Formal selection of cementitious grout for disposal of selected liquid wastes in near-surface vaults was made in the *Hanford Waste Management Plan* (DOE-RL 1983). This selection was strongly influenced by the generally favorable Oak Ridge National Laboratory (ORNL) site grout hydrofracture disposal experience and by the Savannah River Laboratory (SRL) site evaluation and selection of a grout waste form for the disposal of certain aqueous LLW salt solutions. This selection was supported by an independent, comprehensive evaluation performed by Hanford Site scientists and engineers in 1980. This evaluation showed grout to be preferred over other known forms for immobilization and bulk disposal of Hanford Site liquid LLWs (RHO 1980).

The grout formulation process involves waste sampling, characterization, and product testing to ensure that the grout will meet strength and leachability criteria.

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5.1 REGULATORY CHANGES AFFECTING GROUT

In September 1991, Ecology enacted controls for new sources of toxic air pollutants, requiring a Notice of Construction to be submitted prior to the addition or significant modification of an atmospheric source emitting a toxic pollutant.

In November 1991, the EPA gave advance notice of proposed ruling on toxicity characteristic wastes, which will necessitate further land disposal restriction compliance measures for the GTF.

In January 1991, the EPA published the final rule for liners and leak detection systems for land disposal units. Procedural and technical standards in this ruling have all been met, although language to demonstrate equivalency to this rule had to be added to Part B of the GTF dangerous waste permit application.

5.2 STATUS OF ACTIVITIES IN PROGRESS

GTF Dangerous Waste Permit Application. The GTF permit application is nearly complete; the only unresolved issue is the vault hydrogen mitigation issue. Revision 2 of the permit application is scheduled to be issued in July 1992.

Final Safety Analysis Report. The Final Safety Analysis Report (FSAR) is being prepared for review by the Westinghouse Hanford Safety and Environmental Advisory Council (Tank Waste Disposal Subcouncil). It is expected that the FSAR will be submitted to U.S. Department of Energy, Richland Field Office (RL) in July 1992.

Performance Assessment. The U.S. Department of Energy-Headquarters (DOE-HQ) Performance Assessment Peer Review Panel did not approve the draft Performance Assessment (PA) plan for the GTF. Resolution of comments will require a significant effort, including further testing, modeling, and text revision. Approval of the PA is currently the critical path item for restart of the GTF.

Grout Reformulation. Grout reformulation has been necessary to resolve the issues of heat generation and/or poor wasteform properties in earlier formulations. The ORNL conducted a mixture experiment to determine suitable dry blends for solidification of tank 106-AN waste. Further testing was conducted at PNL. A team of ORNL, PNL, and Westinghouse Hanford scientists has chosen a formulation for grouting waste in a pilot plant run to be conducted in April 1992.

DST Waste Sampling. Characterization results were issued for tank 106-AN and tank 101-AW. The contents of these tanks will be solidified in the next three grout campaigns.

No grout-candidate DST sampling was conducted during the past year. Documentation for sampling candidate tanks 105-AP and 106-AP has been prepared and approved. Documentation for sampling feed tank 102-AP is being prepared. Sampling will be conducted after transfer of tank 106-AN contents. The sampling of candidate tanks 104-AN and 105-AN is desirable but may be difficult to achieve because of safety concerns due to hydrogen generation in these tanks.

Vault Construction. Cover panels have been installed on four vaults (102 through 105). The diffusion barrier has been installed to the level of the cover panels. The vaults will be completed during FY 1992 by Kaiser Engineers Hanford.

Core Sampling. Core sampling of the phosphate/sulfate waste (PSW) vault was completed in March 1992. Laboratory testing and reporting will be completed in 1993.

Cold-Cap Formulation. The report on selection of a cold-cap formulation for the PSW vault by the U.S. Army Corps of Engineers is expected in April 1992 and will be reviewed during FY 1992. The PSW vault is expected to be cold-capped in FY 1993.

Vault Equipment. The second portable instrument house will be delivered in April 1992. Pumps to remove the excess liquid are being procured. Design on the exhauster for vaults 102 and 103 has begun and exhauster risers have been installed.

5.3 NEW ACTIVITIES

Quality Verification. Design and single-use specifications for the Hanford Mobile Solidified Low-Level Radioactive Waste Sampling Unit are currently in the approval cycle. A purchase requisition for fabrication of this truck-mounted grout coring unit will be completed in May 1992.

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Research and nondestructive testing continues to be aggressively pursued. Approval of proposals for ultrasonic testing of grout and for research, design, and testing of in situ electrochemical characterization techniques is also actively pursued.

Vault Hydrogen Issues. Significant resources are being utilized for investigation and mitigation of grout vault hydrogen issues. The three major areas of concern are; buildup of hydrogen gas in the vault vapor space; buildup of hydrogen gas in the leachate void space (in the 30-yr time frame); and possible pressurization of the vault after it is sealed. There also appears to be a small potential for the buildup of flammable concentrations of hydrogen in the vapor space of the leachate system. Administrative controls, additional vault equipment, and/or vault design changes may be necessary.

5.4 WASTE GENERATION

The GTF did not operate during the time period covered by this report. A total of 0.20 m^3 (7.4 ft^3) of mixed waste was generated due to maintenance activities and PSW core sampling.

5.5 WASTE MINIMIZATION ACTIVITIES

A cover was installed on the grout Processing Facility/Mixer Module to prevent precipitation from entering the liquid collection tank/mixer module. Formerly this liquid had to be pumped to tank farms. An estimated 4.54 m^3 (1,200 gal) of DST wastes per year are now eliminated.

Products also are being tested to replace aerosols and regulated solvent-based products currently being used.

Substitution of propylene glycol for ethylene glycol in the chiller system for makeup air at the Grout Processing Facility is planned for FY 1992.

5.6 ESTIMATE OF PLANNED WORK ACTIVITIES FOR FISCAL YEAR 1993

Much of the planned work effort for FY 1993 will be focused upon the completion of major ongoing tasks; i.e., approval of the Part B permit, FSAR, and Performance Assessment, Readiness Review, and the resolution of the hydrogen issues.

PNL will operate a 1/4 scale grout pilot plant in April and May 1992. Simulant 106-AN waste will be mixed with a selected blend of dry materials. The grout produced will be cured in two different molds. The first mold will be used to determine the affect of varying curing temperatures of the grout product. The second mold will be used to determine the effectiveness of forced ventilation heat removal from the grout.

A new waste minimization plan will be developed. The plan will comply with DOE Orders 5400.1 (DOE 1988b), 5400.3 (DOE 1989), and 5820.2A (DOE 1984).

The *Washington Administrative Code* (WAC) 173-307 will be used as guidance for the development of this plan. Many of the waste streams will have to be estimated since the GTF is not currently in operation.

The core samples taken from the PSW vault will be analyzed and a test report will be generated.

The contents of tank 106-AN will be transferred into feed tank 102-AP, which will then be sampled and characterized. Small grout samples will be made with the radioactive waste to ensure that the grouted waste will meet the processing and wasteform criteria (Riebling and Fadeff 1991).

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6.0 WASTE FORM QUALIFICATION ACTIVITIES

6.1 INTRODUCTION

This section on waste form qualification activities will provide pertinent background information and FY 1992 program updates on the following topics related to the remediation of HLW stored at the Hanford Site:

- Waste form selection
- Hanford Waste Vitrification Project.

6.2 WASTE FORM SELECTION

The DOE has initiated a remediation program for the disposal of high-level nuclear wastes currently stored in tanks at several DOE sites within the U.S. To date, the U.S. program has selected borosilicate glass as the waste form of choice for use in disposing of all, or at least a significant part, of such wastes that are stored at three of these sites; the Savannah River Site in South Carolina, the West Valley Demonstration Project in New York, and the Hanford Site in the state of Washington.

For the Hanford Site, DOE decided to use borosilicate glass as the waste form for the disposal of the HLW currently stored in DSTs (DOE 1988a). Although HLW is also stored in SSTs on the Hanford Site, final selection of the waste form for the HLW stored in SSTs had not been made during this reporting period. However, it should be noted that borosilicate glass is also one of the leading waste form choices for SST HLWs. The TWRS program for the Hanford Site is currently being rebaselined. One of the major objectives being addressed is that of creating a fully integrated program for the overall remediation of both DSTs and SSTs HLW. The TWRS program rebaselining is to be completed by March 1993.

6.3 HANFORD WASTE VITRIFICATION PLANT PROJECT WASTE FORM QUALIFICATION ACTIVITIES

The following waste form qualification activities are important to support the HWVP project:

- Waste (form product) acceptance specification

- Hanford Waste Vitrification Plant project compliance plan with the waste acceptance specifications.

In 1990, the DOE repository program in the Office of Civilian Radioactive Waste Management (OCRWM) revised the acceptance specifications for a HLW form product consisting of borosilicate glass and the HLW constituents placed in a stainless steel canister. The June 1991 draft of the Waste Acceptance Preliminary Specifications (WAPS) entered a RL high-level review process late in FY 1991 that continued during FY 1992. The DOE Office of Environmental Management (EM) is presently awaiting formal notification on the status of

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WAPS and its attendant schedule. However, at the written request of RL, the HWVP project has been initiated using the June 1991 draft of the WAPS to support project planning.

During FY 1992, the HWVP project prepared a plan that presented the waste form qualification activities and the hierarchy of strategies being used to comply with waste form qualification requirements. In addition to the WFQ program plan, the HWVP project prepared an initial draft for internal project review of the Waste Compliance Plan. Collectively these documents, when completed, will describe the activities that must be accomplished to ensure that the HWVP will produce a product that meets all of the acceptance specifications of the WAPS.

In support of the general design requirements for the HWVP, which include WFQ requirements derived from those for the WAPS, testing and analysis work continued on the development of algorithms that relate the glass composition to its physical and chemical properties. This information is then used to define the acceptable composition range to satisfy both the WFQ and the production requirements (e.g., production rate, waste loading fraction, etc.) for each waste feed option. The above information is also needed to conduct assessments on waste feed processability. A revision of the waste feed processability assessment for DSTs will be completed during FY 1992.

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7.0 ALTERNATIVE TREATMENT/DISPOSAL TECHNOLOGY

This section documents the studies, activities, and issues which occurred in this area over the period of March 1, 1991, through February 29, 1992.

The Underground Storage Tank Integrated Demonstration, funded by the DOE OTD, will examine alternative technologies and technology systems for waste treatment and disposal as part of the overall remediation of DOE mixed waste tanks.

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8.0 SAFETY ISSUES

Section 3137, "Safety Measures for Waste Tanks at Hanford Nuclear Reservation," of Public Law 101-510 (Wyden Amendment), addresses safety issues concerning the handling of high-level nuclear waste in storage tanks at Hanford Site.

Section 3137 specifically addresses the issues concerning the Hanford Site waste tanks by directing that the Secretary of Energy take the following actions:

- Identify those tanks that "...may have a serious potential for release of high-level waste due to uncontrolled increases in temperature or pressure... ."
- Ensure that "...continuous monitoring to detect a release or excessive temperature or pressure..." is being carried out.
- "...develop action plans to respond to excessive temperature or pressure or a release from any tank identified... ."
- Restrict additions of high-level nuclear wastes to the identified tanks unless no safer alternative exists or the serious potential for a release of high-level nuclear waste is no longer a threat.

Compliance activities associated with Section 3137 have resulted in the identification of fifty-three tanks that "...may have a serious potential for release of high-level waste due to uncontrolled increases in temperature or pressure."

More tanks may be identified for addition to the list as characterization of the tank contents continues. However, some tanks may be recommended for removal from the list based on a detailed characterization of the contents, to substantiate better definition and assessment of the risk. Instrumentation to provide additional monitoring for the identified tanks, as well as other improvements to increase monitoring capabilities throughout the tank farms, are being developed on an expedited basis and are being implemented readily.

Action plans to respond as appropriate and technically feasible to excessive temperature or pressure or a release from ferrocyanide tanks are in place (Cash and Thurman 1991). The response plans for the remaining identified tanks are being prepared. As upgraded monitoring capabilities are implemented, these plans will be modified accordingly, where applicable.

The SSTs that have been inactive are isolated on an interim basis and all transfer lines that could transfer high-level nuclear waste have been physically isolated from the tanks. For those SSTs that have not been isolated, however, the associated transfer lines into the tanks have been physically isolated from the tanks. Modifications to the Safety Analysis Report (SAR), requiring RL approval, would be needed before reconnections. The DSTs are considered active and are not physically isolated. The five DSTs identified in accordance with Section 3137 are excluded from becoming active receiver tanks. Blocking valves, between the identified DSTs and the transfer

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lines, are closed and locked and tagged in accordance with approved procedures to ensure that no transfers to these tanks can take place.

8.1 DESCRIPTION OF TANK WASTE SAFETY ISSUES

This section provides an updated overview of five major safety issues associated with SSTs and DSTs and their potential impact on waste treatment. The first four safety issues have already been identified as Priority I. Priority I is defined as issues and/or situations that contain most of the necessary conditions that could lead to worker (onsite) or offsite radiation exposure through an uncontrolled release of fission products. Issues of concern to potential treatment strategies include the following:

- Flammable gas generation in tank 101-SY and other tanks
- Potential explosive mixtures of ferrocyanide in tanks
- Potential organic-nitrate reactions in tanks
- Continued cooling requirements for high heat generation in tank 106-C
- Criticality concerns in selected waste tanks.

Safety issues focus on the Waste Tank Safety Program to ensure the safety of the SST and DST systems until appropriate treatment and disposal of their contents can be implemented. To ensure interim safety, extensive administrative and technical controls are maintained for the safety-issue related tanks identified in Table 8-1. A broad-based peer review of all planning and safety documentation by high-level oversight groups appointed by DOE-HQ is also being conducted. A high-level waste tanks task force and a high-level waste tanks advisory panel at the DOE in the Office of Environmental Restoration and Waste Management have been established. Together with the Hanford Site staff they will ensure that the Hanford Site corrective action programs are technically adequate, have the proper priority, and are on an expeditious schedule for resolution. In addition, DOE approval of all actions relating to those tanks containing flammable gases and/or ferrocyanide compounds is required.

The hazardous characteristics of the existing SST and DST wastes, leading to their identification and control, currently are being evaluated on the basis of pertinent chemical literature, expert peer judgment, and limited sampling data. Mitigating factors, such as moisture content, presence of relatively inert diluents (e.g., sodium carbonate, sodium aluminate, and/or sodium phosphate) and any other conditions that could reduce reactivity of the wastes, are being analyzed.

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Table 8-1. Safety Issue Tanks.

Flammable-gas generating	Ferrocyanide	Organic	High heat
Single-shell	Single-shell	Single-shell	Single-shell
101-A	102-BX	103-C	106-C
101-AX	106-BX	103-B	
103-AX	110-BX	105-TX	
102-S	111-BX	118-TX	
111-S	101-BY	102-S	
112-S	103-BY	106-SX	
101-SX	104-BY	106-U	
102-SX	105-BY	106-U	
103-SX	106-BY		
104-SX	107-BY		
105-SX	108-BY		
106-SX	110-BY		
109-SX	111-BY		
110-T	112-BY		
103-U	108-C		
105-U	109-C		
108-U	111-C		
109-U	112-C		
	101-T		
Double-shell	107-T		
	118-TX		
103-AN	101-TY		
104-AN	103-TY		
105-AN	104-TY		
101-SY			
103-SY			

NOTE: The underlined tanks also appear on either the flammable gases or ferrocyanide lists.

Scenarios of significant concern associated with waste in tanks include the following.

- Potential for ignition of flammable gases, such as hydrogen-air and hydrogen-nitrous oxide.
- Potential for ignition of organic-nitrate mixtures initiated by the radiolytic and/or chemical heating of dry saltcake.
- Potential for ignition of ferrocyanide-nitrate mixtures initiated by the radiolytic and/or chemical heating of dry saltcake.
- Potential for tank leakage causing contaminant release to the environment while simultaneously meeting a requirement for addition of cooling water to tank 106-C to maintain its structural integrity.

Administrative and technical controls are implemented to restrict activities which could cause any abnormal, undesirable events. For example, pumping of interstitial liquid from tanks containing ferrocyanide has been eliminated to maintain present in-tank chemical stability. Nonsparking tools and use of electrical bonding techniques on tank instrumentation are also mandated. Normal activities for tanks at issue are limited to surveillance. Preparation of special safety analysis documents, which are extensively reviewed by the aforementioned peer groups, are prepared for all in-tank work activities.

Comprehensive monitoring, characterization, and attendant applied research activities have been initiated to support resolution of the current key issues and any future safety concerns related to potential waste incompatibilities or actions from planned treatment and disposal of selected tank wastes. Such efforts will also provide a sound basis for near-term remediation of tanks and will aid in defining the envelope of safety to support the disposal of all tank wastes at the Hanford Site. A plan to implement remediation of waste tank safety issues at the Hanford Site has been prepared (Wilson and Reep 1991).

8.2 FLAMMABLE GAS GENERATING TANKS

One DST, tank 101-SY, generates, stores, and periodically releases significant quantities of flammable gases, primarily hydrogen and nitrous oxide. Tank 101-SY contains a mixture of DSS and CC, which is a high organic containing waste. If a spark were to be present, this gas could ignite and burn, potentially causing filters in the vent system to fail with resulting spread of contamination. Tank 101-SY was previously identified as an unreviewed safety question.

Flammable gas generation in tank 101-SY is a top priority waste tank safety issue at the Hanford Site because peak concentrations above the lower flammability limit (LFL) for hydrogen occur periodically. The tank has vented up to an estimated 340 m^3 (12,000 ft^3) of gas (containing about 38 percent ± 4 percent hydrogen and 32 percent ± 4 percent nitrous oxide). The venting is a function of temperature or gas bubble instability, which causes the gas generated deep within the tank to move up to the top of the tank. The gas then vents into the dome space in the top of the tank and is removed through a filtered ventilation system. Such venting of gases is expected to keep recurring until some form of remediation is taken. During the episodic venting, the tank is sometimes brought to positive pressure for a few minutes by the rapidity of the gas release. In addition, it is likely that a greater than LFL concentration exists at times in the waste tank. If an ignition source were present during these periods, a hydrogen burn or explosion could occur causing harmful radiation exposure to onsite and offsite personnel. However, the small pressurizations that have occurred to date have not resulted in any contamination spread associated with the event.

In addition to tank 101-SY there are 22 other tanks (four DSTs and 18 SSTs) also suspected of potentially containing smaller accumulations of hydrogen or other flammable gases. Evidence of venting, surface level behavior, and knowledge of the other tank contents suggests a lower likelihood of potentially dangerous gas concentrations in these other tanks.

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The goal of the flammable gas study program is to gain sufficient understanding by peers of the causes and patterns of gas generation to allow DOE to initiate mitigation or remediation of the potentially hazardous situation.

In general, all actions proposed to gain information (characterization) or enhance safety (e.g., added ventilation) require orderly and detailed safety assessment of their safety implications.

Options currently being considered for interim remediation include (after sampling the waste) (1) diluting and mixing the waste, (2) transferring the waste to other DSTs and then diluting and mixing it in the affected tanks, (3) increasing ventilation to remove gases faster, (4) stirring and/or mixing to release gas bubbles, and (5) heating or ultrasonic bubble breakup.

Planning for characterization, mitigation, and interim remediation (as appropriate) of all 23 tanks that generate hydrogen or other combustible gases has been initiated. Plans include sampling each tank to support safe mitigation.

At this time, there are insufficient data and analyses to permit selection of any remedial method. All concepts will be pursued in parallel with the waste characterization and laboratory studies. As more adequate information becomes available on the nature of the waste and the mechanism for gas production and its release, it will be possible to focus on future remedial actions.

The remedial approach will address the episodic releases of hydrogen, nitrous oxide, and nitrogen from tank 101-SY at approximately 100 days periodicity. Near-term interim remediation efforts will be directed at eliminating cyclic release of gases, thereby allowing for a continuous release at gas concentrations well below safety limits. Methods being considered are the following:

- Transfer of partial tank contents to another tank
- Dilution of tank contents
- Mixing of tank contents (pumping, stirring, ultrasonic agitation)
- A combination of transferring and diluting
- A combination of transferring and mixing
- A combination of transferring, diluting, and mixing
- Ultrasonic degasification.

Currently, there is marginal extra tank capacity available at the Hanford Site. Development of remedial actions may require the construction of additional tanks. However, tank 103-SY has approximately 950 m^3 (250,000 gal) of free space, and the existing waste in tank 103-SY may be compatible for mixing and dilution.

Long-term interim remedial actions will be directed at slowing down or stopping the process that produces the gases. Methods being considered include chemical and/or thermal treatment of the waste. These methods will require detailed chemical and engineering analysis and development.

8.2.1 Action Plan

A program has been initiated to develop and implement a solution to this safety issue. The major objectives of this program are the following:

- Ensure that no accidents occur during the continued operation of these tanks
- Upgrade the monitoring capability of the tanks
- Resolve tank safety.

8.2.1.1 Safety Studies. Detailed safety studies have been implemented and are continuing for the tanks containing flammable gas. These studies and analyses are focused in two areas:

- Safety assessments of in-tank operations in accordance with DOE Order 5480.5 (DOE 1986a) (DOE 1988a)
- Safety evaluation and accident risk analysis.

8.2.1.2 Waste Characterization and Modeling. Determination of reaction mechanisms require a detailed characterization of waste samples obtained through core drilling of the tanks. Postulated mechanisms are being evaluated through laboratory studies on synthetic and actual waste materials. Results will be used to develop a model for the thermo-physical and chemical behavior of the waste in the tank. This information will then be used to formulate both interim and final remedial actions.

8.2.1.3 Upgrade Tank Monitoring. To ensure safe operation of the tanks, it is necessary to provide accurate and reliable monitoring of the temperature, pressure, gas flow, gas composition, and surface level of the tank contents. New instrument trees are being designed and constructed to replace old, outdated control and instrumentation for 23 of these tanks. Tank 101-SY will be the first tank to have a new instrument tree installed. In addition to these trees, other monitoring equipment to measure temperature, gas flow, pressure, and humidity will be located in the exhaust system. Monitoring equipment, such as television cameras, infrared scanners, and radar level gauges, is also being implemented. Because the release of gases in the tanks can cause an increase in pressure in the tank dome space, upgraded ventilation systems will be developed as needed.

8.2.1.4 Upgrade Tank Ventilation. Currently, a study has been completed to define the requirements for new ventilation systems.

Accordingly, the system is required to maintain a "negative" tank pressure at all times and must provide for minimizing the amount of flammable gas mixtures that would exceed the LFL.

8.2.1.5 Interim Remediation. Although a number of potential remedial actions have been proposed, detailed engineering studies will be required to select the most effective and timely approach for in-tank processing. For interim remediation, the criteria will be established and the proposed concepts will then be evaluated against the criteria. One or more concepts will be selected

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for detailed study. Upon final selection of a concept, it will be set up as a project with the normal elements of design, fabrication, development procedures, training, and safety reviews. This activity will also include preparation of appropriate *National Environmental Policy Act of 1969* (NEPA) documentation. However, no salt well pumping is currently being conducted because of a concern that the temperature may increase, causing an exothermic reaction.

8.3 TANKS CONTAINING FERROCYANIDE

Ferrocyanide tanks were selected as the second major issue within the Priority I class of safety issues. Concentrations and distributions of ferrocyanide and nitrate and/or nitrite materials within the tanks could lead to a potential explosion, if tank contents were allowed to heat up or if an uncontrolled exothermic reaction could occur. Currently, twenty-four SSTs contain insoluble ferrocyanide salts in quantities greater than 1,000 g-mol (465 lb) mixed in a sodium nitrate/sodium nitrite matrix. This mass represents the threshold quantity. A total of approximately 140 metric tons (310,000 lb) of ferrocyanide is contained within these tanks. The ferrocyanide concentration ranges from 1,000 g-mole (calculated as the ferrocyanide anion) to a maximum of approximately 200,000 g-mole (93,000 lb) in tank 104-BY. If subjected to high temperatures, above 285 °C (545 °F), these materials could become explosive. Some of these tanks also may contain quantities of organic materials that cause exothermic reactions at a lower end of the temperature range, i.e., 180 to 200 °C (356 to 392 °F). However, there is a low probability for any heating mechanism to occur. Based on available information (as of November 1991) which has been reviewed and analyzed with regard to tank storage safety (Postma et al. 1991), it is concluded that most, if not all, of the tank waste is nonreactive in its present form. Nevertheless, additional information is needed to confirm these initial conclusions. Surveillance and control systems must be developed to safeguard against explosion and/or fire in these tanks as they contain significant quantities of sodium nitrate, sodium nitrite, silicates, aluminates, hydroxides, phosphates, sulfates, carbonates, uranium, copper, calcium, and fission products from the processing of irradiated fuel.

In summary, concentrated ferrocyanide nitrate/nitrite chemical combinations can undergo an oxidation-reduction reaction; laboratory tests have demonstrated that these chemicals, when dry and relatively pure, can react exothermally. On the other hand, it has been shown that the ferrocyanide-nitrate/nitrite reaction cannot propagate through wastes if the reactants are diluted by inert chemicals and/or water. For a specific waste storage tank, the key parameters that would govern waste reactivity are:

- The mass of ferrocyanide (inventory)
- The proportion of diluents present (concentration)
- The proportion of water present (percent moisture)
- The temperature of the stored wastes.

A better knowledge of these parameters is needed to confirm potential waste reactivity, since previous assessments of ferrocyanide nitrate/nitrite reactions have given a mixed picture. Some assessments indicated that a

significant reaction under storage conditions was not possible; in others, explosive reactions were postulated. Therefore, analyses of actual waste samples are needed to clarify these differences.

A recent study to determine an understanding of the safety of storing high-level waste containing ferrocyanide at the Hanford Site (Postma et al. 1991) presented the following preliminary conclusions about waste in the tanks.

- Ferrocyanide concentrations in most tanks are too diluted by inert chemicals and water to support a propagating reaction.
- Tank contents are different from each other; therefore, the tanks must be treated individually in risk assessments.

This study also presented the following conclusions related to continued in situ storage.

- Dryout of wastes by evaporation of water into dry air flowing through the head space should be prevented.
- Criteria for safe storage should be developed to guide tank management and surveillance operations. The key parameters are moisture content and temperature.
- Tanks should be monitored (temperature, moisture) to verify that safe storage conditions do not deteriorate with time.
- Control equipment should be installed to permit a quick response in the event that moisture or temperatures deviate from specified safe limits.
- Emergency preparedness procedures should be reevaluated with respect to the above conditions.

Ferrocyanide tanks were identified as an unreviewed safety question because it is not known whether concentrations and distribution of ferrocyanide and nitrate-nitrite materials in the tanks would allow an uncontrolled exothermic reaction or explosion if tank contents were allowed to heat up. Although the measured tank temperatures are far below the temperature required to cause an exothermic reaction, the consequences of an event could be at a level potentially exceeding the safety envelope defined in the Environmental Impact Statement (EIS) (DOE 1987) (GAO 1990).

The probability of a ferrocyanide explosion during storage is considered very low because currently measured maximum temperatures in the ferrocyanide tanks [57 °C (135 °F)] falls significantly below the lowest threshold temperature 180 to 200 °C (356 to 392 °F) for ferrocyanide nitrate-nitrite reactions found in the laboratory. Administrative controls are in place to ensure that conditions are avoided that could lead to creation of temperature rises in the tank. Efforts are focused on enhancing monitoring capability, characterizing tank 104-BY, and gaining information on the mechanism and propagation and radionuclide release characteristics of a ferrocyanide explosion.

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A recent review (Babad and Deichman 1991a, 1991b) of the practice of pumping liquid out of SSTs into the soil to avoid potential leakage of radioactive and hazardous materials ascertained that additional analysis of this practice for the ferrocyanide tanks is needed. For tanks that contain large quantities of ignitable materials (tanks containing ferrocyanide and organics) such pumping has been discontinued until safety evaluations of liquid removal can be completed. Verifying that the interstitial and supernatant liquid can be safely removed from tanks containing ferrocyanide is a key part of meeting the agreements set forth in the Tri-Party Agreement (Ecology et al. 1990).

8.4 TANKS CONTAINING ORGANIC WASTE

High concentrations of organic compounds have been found (from tank transfer, flow sheet records, and limited analytical data) in eight SSTs that contain organic chemical salts, and other hydrocarbons such as hexone, esters (tributylphosphate), and NPH at concentrations believed to be greater than 10 mol percent sodium acetate equivalent, mixed in a sodium nitrate-sodium nitrite matrix. Such a mixture is potentially reactive at temperatures above 180 °C (356 °F). Thus, significant overheating of the tank possibly could damage the tank and lead to releases of radioactive materials to the environment. Two of the hydrogen tanks (102-S and 106-SX) and one of the ferrocyanide tanks (118-TX) also appear on the organic list.

Concentrations of organics may be present in some tanks that could cause an exothermic reaction given a sufficient driving force, such as high temperature. However, the difference between ignition temperatures and actual tank content temperatures measured, as discussed previously for the ferrocyanide tanks, is large enough (80 °C vs. 57 °C) that the probability of such a reaction is considered very low. The consequences of the postulated reaction is about the same as that for some scenarios for an explosion in a "burping" hydrogen tank.

The primary points of concern with the tanks containing organic compounds include assessing the following:

- The degree of potential for ignition of flammable gases such as air-organic vapor mixtures
- The degree of potential for ignition of organic-nitrate and/or organic-nitrite mixtures
- The generation of toxic vapors
- The degree of potential for ignition of organic-nitrate and/or organic-nitrite mixtures being initiated from radiolytic or chemical heating of the saltcake mixture
- The verification that existing concentrations in the tanks are safe to store
- The determination that removal and treatment of the waste is required to ensure safe storage until final disposal.

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Future plans and studies include safety analyses of all applicable SSTs and DSTs and their contents to identify those tanks that contain unsafe amounts of organics. This will include dose consequence analysis and probabilistic risk assessments. This activity also provides for the overall safety management and control of the activities and systems associated with the tanks containing significant quantities of organic chemicals. Tanks that contain possible combustible or explosive reactants will also be analyzed.

Future activities also include a detailed evaluation of the available records to determine whether other tanks contain a high organic content.

Through laboratory studies, work is also planned to more accurately determine the initiation point for organic-nitrate and/or nitrite exothermic reactions that can become unsafe. Although tank temperatures appear to be stable or decreasing, additional work is planned to ensure that temperatures measured at various locations in the tank are representative of the entire tank contents.

Future efforts also include tank sampling and laboratory analysis to gain a better understanding of the chemical mixtures present in the tanks. From this knowledge, mathematical models will be developed for evaluating and postulating chemical reactions and to determine the potential for an unsafe reaction. These reactions will be studied in detail to determine safety requirements for the tanks.

Activities will also be initiated to upgrade the instrumentation for tank monitoring and to upgrade existing tank ventilation systems, where necessary. These projects will ensure adequate airflow, filtration, and exhaust monitoring to eliminate any safety concerns associated with organics generating gas in the waste tanks.

Interim remediation, stabilization, and potential final treatment and remediation need to be identified and developed so that strategies can be developed and safely implemented. The strategies will include the development of criteria, alternatives, and the selection of alternatives for further development. A preferred alternative is planned for implementation after NEPA evaluation. Currently, no saltwell pumping is being conducted because the tank temperature may increase, causing an exothermic reaction.

8.5 HIGH-HEAT TANK

One tank requires periodic addition of water and forced air ventilation to maintain its temperature within the permissible limits determined by structural considerations. Tank 106-C was identified as a safety concern.

Single-shell tank 106-C is a 2.0 ML (530,000 gal) tank located in the C Tank Farm in the 200 East Area. This tank has been used for radioactive waste storage since mid-1947 and currently contains about 950,000 L (250,000 gal) of waste. During the late 1960's, a program to recover strontium and cesium from aging stored waste in the A and AX Tank Farms was instituted at the Hanford Site. Sludge washing/decanting steps in this process inadvertently transferred heat-generating strontium-rich sludge to tank 106-C. However, the tank integrity currently is considered sound.

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Since mid-1971, water has been added periodically to tank 106-C to keep the sludge wet and to promote heat transfer by evaporation to the vapor space. If tank 106-C leaks, the need for cooling water would remain. Interstitial liquid could not be removed to sufficiently stop leakage to the environment. The consequences of this phenomena would allow a localized leak of contamination into the soil. If the current methods of cooling tank 106-C are stopped, the sludge temperatures may exceed established limits and may cause tank structural damage, leading to dome collapse and possibly an unacceptable radioactive release to the environment.

A Tri-Party Agreement (Ecology et al. 1990) milestone has been established to interim stabilizing tank 106-C by removing most of the interstitial liquids by September 1996. Accordingly, any process that periodically adds water to the tank will be eliminated. Studies indicate that the heat-generation rate of 43.96 kw (150,000 Btu/h) is too large to eliminate the current means of cooling of tank 106-C without providing an alternative.

There are three options that can be used to maintain the heat within tank 106-C at a level that will be acceptable from a structural point of view.

- Continue to add cooling water periodically, which could result in environmental releases should the tank leak
- Retrieve or partially retrieve the material from tank 106-C and dilute or treat it to remove the high heat source
- Provide a mechanical means of controlling the heat within the sludge.

The first option is undesirable because water additions to the tank would provide a means for releasing additional radionuclides to the soil should the tank leak.

The second option has been studied previously, and retrieval in itself can technically be accomplished; currently this is the preferred alternative (Esvelt 1990). The problem lies in the lack of retrieval systems and in the lack of existing tank space.

The third option would require installing heat exchangers or ventilators within the tank to ensure that the sludge could be maintained at the acceptable temperature level. The last two alternatives offer the greatest potential to pursue until a definitive cost advantage of one over the other emerges.

8.6 CRITICALITY CONCERNS IN SELECTED TANKS

Boundary limits for the amount of radionuclides in the DSTs have been set to ensure that a criticality reaction cannot occur (Halgren 1990). The

plutonium concentration is checked by sampling and analysis prior to discharge in the tanks by the generator of the waste. There are two limits associated with plutonium as follows.

- The total plutonium content per tank transfer of waste must be less than 200 g.
- The maximum total plutonium content of a tank must be less than 0.013 g/L (0.05 g/gal) of waste.

However, the above limit of fissile materials content for SSTs in the recent waste characterization plan has not been specified. Currently, there is no precise accounting of fissile materials for SSTs. Initially, a reevaluation of historical waste transfer records is needed to assess the safety implication for SSTs. Work has been initiated to resolve these tank waste safety issues.

8.7 POTENTIAL IMPACT ON TREATMENT

Extensive requirements for peer review and associated approvals for any intrusive action in listed tanks (Table 8-1) could impact both cost and schedule associated with treatment of tank wastes. In addition, the existence of potentially incompatible mixtures of chemicals in the tanks will impose temperature limitations on the retrieval operations and might require modification of pretreatment flowsheets to either destroy reactive components or to require separation of fuel from oxidizers.

The waste tank safety program has recommended that temperature limitations be imposed on all aspects of retrieval to limit edge-of-tool temperatures to below 150 °C (302 °F). As work progresses, the program will determine the degree to which the listed tanks do indeed pose a near-term or inherent safety problem with respect to safe storage. Many of the mitigation and/or remediation strategies that are being evaluated for tank 101-SY should be broadly applicable to other tank wastes. The focus for the ferrocyanide program is more clearly defined as an envelope of risk for an explosion of heated tank wastes. The organic program planning effort is continuing and remediation alternatives currently are being evaluated. Remediation alternatives for tank 106-C are also being evaluated.

The safety program is actively pursuing both the SST and DST treatment and disposal programs to ensure that all engineering approaches accommodate the potential risk associated with the watch list tanks.

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APPENDIX A

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LIST OF TERMS

ALC	airlift circulator
CAW	current acid waste
CY	calendar year
D&D	decontamination and decommissioning
DOE	U.S. Department of Energy
DOE-HQ	U.S. Department of Energy, Headquarters
DMF	Dry Materials Facility
DSSF	double-shell slurry feed
DST	double-shell tank
Ecology	Washington State Department of Ecology
EPA	U.S. Environmental Protection Agency
FFTF	Fast Flux Test Facility
FY	fiscal year
GDF	Grouted Waste Disposal Facility
GPF	Grout Processing Facility
GTF	Grout Treatment Facility
HVAC	heating, ventilation, and air conditioning
HWVP	Hanford Waste Vitrification Plant
IEM	Interim Examination and Maintenance (Cell)
LLW	low-level waste
MAP	mixed activation products
MASF	Maintenance and Storage Facility
MFP	mixed fission products
NCAW	neutralized current acid waste
NCRW	neutralized cladding removal waste
NZAW	neutralized zirflex acid waste
OA	outside air
OSR	Operational Safety Report
PFP	Plutonium Finishing Plant
PNL	Pacific Northwest Laboratory
PRP	Plutonium Reclamation Facility
PUREX	Plutonium/Uranium Extraction (Plant)
RL	U.S. Department of Energy, Richland Field Office
RLWS	Radioactive Liquid Waste System
RMC	Remote Mechanical C-Line
RMW	radioactive mixed waste
SST	single-shell tank
SWL	saltwell liquor
TRU	transuranic (waste)
UNH	uranyl nitrate hexahydrate
WAC	*Washington Administrative Code*
WESF	Waste Encapsulation and Storage Facility
Westinghouse Hanford	Westinghouse Hanford Company

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A.1.0 100 N AREA

This section documents the studies, activities, and issues which occurred in the 100 N Area during the period of March 1, 1991, through February 29, 1992.

A.1.1 INTRODUCTION

The principal facility in the 100 N Area is the dual-purpose N Reactor, which was designed to produce special nuclear materials and steam for generating electricity. Support facilities for N Reactor include a water-filled fuel storage basin and decontamination systems for both the reactor and fuel storage basin.

The three primary types of waste generated at this facility during operation are:

- N Reactor decontamination waste
- Ion-exchange regeneration waste
- Sand filter backwash.

Due to the standby status of the N Reactor, no new waste from reactor operations was generated during the period from March 1991 through February 1992.

A.1.2 SUMMARY OF MARCH 1991 THROUGH FEBRUARY 1992 ACTIVITIES

Generation of 136 m^3 (36,000 gal) of Waste. This section traces the processing of the remaining waste stored in the fuel storage basin which would have generated an estimated 136 m^3 (36,000 gal) of waste as mentioned in Section 1.2.2, Appendix A, of the 1990 *Annual Report of Tank Waste Treatability* (Karnesky 1990).

The generation of this waste will not take place for two reasons.

- There is limited 200 Area tank space.

- The need for ion-exchange column use and regeneration has been eliminated because of a reduction of storage basin water radionuclide concentrations experienced since the completion of irradiated-fuel transfers to the K-Basins in December 1989.

A.1.3 STATUS OF 1992 ACTIVITIES IN PROGRESS

A sand filter is used to remove entrained solids from the fuel storage basin water before treatment with ion-exchange during normal operations. The sand filter backwash is primarily an inorganic sludge generated during

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periodic filter flushing to remove accumulated solids. The sand filters at 107-N have been shut down. The system will not be used again until basin cleanup activities commence in the 1994/1995 time period.

A.1.4 CURRENT INVENTORY AND/OR AMOUNTS GENERATED

The regenerative waste tank in 107-N is currently holding 75.7 m^3 (20,000 gal) of sulfate waste that will be shipped to the tank farms in fiscal year (FY) 1993.

A.1.5 WASTE MINIMIZATION ACTIVITIES

No new waste minimization activities are in place.

A.1.6 ESTIMATE OF PLANNED WORK ACTIVITIES FOR 1992

The following activities are planned for 1992.

- 56.8 m^3 (15,000 gal) of liquid wash-down waste is expected from tank cleanout and layup activities.
- The operation of the sand filters mentioned above in Section A.1.3 necessitates backwashes that add to the sludge volume in the backwash settling tank. The sludge hold-up volume is estimated to be 3.8 m^3 (1,000 gal). This sulfate waste also is projected to be shipped in FY 1993, but will require additional liquid for dilution due to the fissile content and high dose rate experienced because of the concentration of radionuclides present in the constituent. The requirement for dilution is estimated to be 340.6 m^3 (90,000 gal).
- N Reactor has received a FY 1991 shutdown order. Therefore, decontamination and decommissioning (D&D) of the inactive production reactors would represent a potential large-scale activity which would then generate an undetermined quantity of decontamination-related waste.

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A.2.0 CURRENT WASTE GENERATORS IN THE 300 AREA

This section documents the studies, activities, and issues which occurred in this area during the period of March 1, 1991 through February 29, 1992.

A.2.1 DESCRIPTION OF FACILITIES AND TYPES OF WASTES GENERATED

In the 300 Area, tank waste is generated in seven different laboratory facilities and transferred to the 340 Waste Handling Facility for shipment to the tank farms for storage, any necessary treatment, and ultimate disposal. Since the 1991 report, two facilities in the 3000 Area (LSL-II and RTL Facility) have generated waste that is being disposed of either via transport directly to the 340 Facility, or transport to the RLWS drain in the 329 Building (300 Area).

Descriptions of the seven individual laboratory facilities; the 3000 Area facilities, the 340 Facility, and their individual waste streams are presented in this section. A composite analysis of the tank waste generated in the 300 and the 3000 Areas is included in the discussion of the 340 Facility.

A.2.1.1 324 Chemical Engineering Laboratory

The 324 Chemical Engineering Laboratory contribution to tank waste is primarily from two groups of shielded hot-cells and their service and operating galleries. Liquid wastes that are produced during the operation of these hot-cell facilities are pumped from vault tanks through the RLWS line to the 340 Facility for temporary storage before transfer by rail tank car to the tank farms. In some cases, wastes are delivered to the 340 Facility in steel drums.

The 324 Chemical Engineering Laboratory's contribution to tank waste for 1991/1992 was considerably lower than the amount generated in 1990. This is because the emphasis was on emptying and transferring the contents of the various tanks. Consequently, there are only small amounts of material remaining in these tanks. The waste streams from the 324 Facility consist mainly of small project waste as follows:

- Volume--189 L/yr (50 gal/yr)
- Chemical composition--mainly water
- Predominant radionuclides--^{137}Cs and ^{90}Sr with mixed fission products (MFP) and mixed activation products (MAP).

A.2.1.2 325 Radiochemistry Laboratory

The 325 Radiochemistry Laboratory is a multipurpose laboratory facility with two different sets of hot-cells and several analytical laboratories. Since 1990, waste volumes have increased in each laboratory area within the

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325 Building complex. This can be attributed to the restart of single-shell tank (SST)/double-shell tank (DST) core characterization activities. Thus, the waste volume may fluctuate depending on tank core characterization priorities.

The hot-cells located in the east wing of the 325 Building (325A) are used to handle highly radioactive materials for a variety of processes and tests. The inorganic waste produced in the cells generally consists of rinse water and dissolved irradiated fuel sample sections. The hot-cells are also used to extrude and blend core samples from the tank farms. A description of the waste that will be generated in the process research hot-cells is as follows:

- Volume--454 L/yr (120 gal/yr)
- Chemical composition--inorganic compounds, water
- Predominant radionuclides--^{144}Ce, ^{60}Co, ^{134}Cs, ^{137}Cs, ^{106}Ru with MFP and MAP.

The hot-cells in the west wing of the 325 Building (325B) are used to prepare fuel component samples, tank cores, and other solid samples for various chemical analyses. The waste that is generated in these hot cells is primarily rinse water. A description of the waste generated in the 325 Building is as follows:

- Volume--4,731 L/yr (1,250 gal/yr)
- Chemical composition--traces of inorganic and organic constituents, water
- Predominant radionuclides--^{144}Ce, ^{60}Co, ^{134}Cs, ^{137}Cs, and ^{106}Ru with MFP and MAP.

The analytical laboratory waste generated in the 325 Building is sent directly to the 340 Facility via the Radioactive Liquid Waste System (RLWS) drains. Most of the waste is generated from fuel rod analysis and tank core characterization. A general description of the waste produced from analytical work is as follows:

- Volume--6,283 L/yr (1,660 gal/yr)
- Chemical composition--inorganic, organic (trace), and analytical waste
- Predominant radionuclides--^{144}Ce, ^{60}Co, ^{134}Cs, ^{137}Cs, and ^{106}Ru with MFP and MAP.

A.2.1.3 326 Materials Technology Laboratory

Most of the work performed in the 326 Materials Technology Laboratory involves the study of metallurgical, chemical, and physical behavior of reactor components and fuel materials. In mid 1991, the RLWS system in the

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326 Building was reopened after being administratively closed. Most of the waste generated in this building was shipped to the Central Waste Complex in steel drums for storage as radioactive mixed waste (RMW). This transfer is performed because the waste usually does not meet the 340 Facility acceptance criteria.

The metallography laboratory, where radioactive waste is generated, is used to prepare metal coupons for survey in an electron microscope. The coupons are prepared by washing them in several different acid baths. A general description of the waste that is generated in this section of the 326 Building in 1990 is as follows:

- Volume--23 L/yr (6 gal/yr)
- Chemical composition--solutions containing trace quantities of perchloric acid, acetic acid, isobutanol, and methanol
- Predominant radionuclides--^{55}Fe, ^{54}Mn, tritium, ^{14}C, ^{63}Ni, ^{60}C, ^{93}Zr, and ^{99}Tc.

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A.2.1.4 327 Post-Irradiation Testing Laboratory

The 327 Postirradiation Testing Laboratory is used for destructive and nondestructive examination of irradiated reactor fuel and structural materials. These examinations and the associated testing are carried out in 12 shielded cells, several of which drain to the 340 Building via the RLWS. The cell drains are filtered to prevent solids from entering the RLWS piping and 340 facility tanks. Most of the waste is generated during grinding and cutting operations that are performed on irradiated fuels and materials, and when the equipment in the cells is cleaned and rinsed. The following is a general description of the waste that is generated by the 327 Laboratory:

- Volume--4,164 L/yr (1,100 gal/yr)
- Chemical composition--water mixed with decontamination materials (traces of detergents, cleaners, surfactants, etc.), low concentrations of isobutanol and methanol
- Predominant radionuclides--^{144}Ce, ^{137}Cs, ^{90}Sr, and ^{60}Co.

A.2.1.5 329 Physics Science Laboratory

The 329 Physics Science Laboratory includes laboratories for radioanalysis and low-level detection and measurement of radioisotopes. Radioactive sources are also manufactured in this laboratory.

The experiments or processes used in the radiochemical portion of the 329 laboratory include dissolution of solids, ion-exchange and precipitation partitioning, and liquid extractions. The following is a description of the waste typically generated in the radiochemistry portion of the 329 Laboratory:

- Volume--549 L/yr (145 gal/yr)

- Chemical composition--nitrate, carbonate, oxalate, sulfate, fluorine, sodium, and ammonia solutions

- Predominant radionuclides--^{241}Am, ^{60}Co, ^{137}Cs, ^{55}Fe, ^{93m}Nb, ^{63}Ni, ^{239}Pu, ^{240}Pu, and ^{90}Sr.

Only a small amount of waste is produced in the low-level detection facility. The following is a general description of the waste produced:

- Volume--3.8 L/yr (1 gal/yr)
- Chemical composition--water
- Predominant radionuclides--^{60}Co, ^{137}Cs, and ^{90}Sr.

A.2.1.6 3720 Building

Several laboratories are housed in the 3720 Building. Activities in the Geochemistry group generates radioactive waste as a result of the study of radioactive grouts and their leachates. The small amount of radioactive waste generated in the 3720 Building (and also the lysimeter site north of the 300 Area) is collected in drums and transported to the 340 Facility where it is added to the accumulation tanks.

A general description of the waste generated in 3720 Building is as follows:

- Volume--151 L/yr (40 gal/yr)

- Chemical composition--varies depending on experiment, mainly groundwater with small amounts of chemical indicators.

- Predominant radionuclides--tritium, ^{60}Co, ^{14}C, ^{99}Tc at or below detection levels.

A.2.1.7 331 Life Sciences Laboratory

The 331 Life Sciences Laboratory is used for a variety of biological and ecological research studies. A small amount of waste generated at the 331 Building was sent to the 340 Facility in 1991 via the RLWS drain in 325 Building. A general description of the waste generated in the 331 Building is as follows:

- Volume--700 L/yr (185 gal/yr)

- Chemical composition--biological liquid wastes containing low concentrations of sodium nitrate, sodium phosphate, and other inorganic compounds

- Predominant radionuclides--tritium, ^{239}Pu, ^{14}C.

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A.2.1.8 3000 Area Facilities

The two facilities in the 3000 Area (LSL-II and RTL) mainly generate liquid scintillation counting waste (non-xylene and/or non-methanol) in support of biological research programs. The wastes are shipped to 329 Building and disposed via the RLWS drain in that facility. A general description of the waste generated in the 3000 Area Facilities is as follows:

- Volume--98 L (26 gal/yr)
- Chemical composition--biological liquid wastes containing non-regulated scintillation cocktail, low concentrations of organic acids
- Predominant radionuclides--tritium, ^{14}C, ^{60}Co, ^{63}Ni.

A.2.1.9 340 Waste Handling Facility

A.2.1.9.1 Description. The 340 Facility is a liquid waste handling facility. Waste is received from Pacific Northwest Laboratory (PNL) via underground pipelines or transported to the 340 Facility in drums and added into the 340 storage tanks. The 340 Facility transfers the waste into 75,700-L (20,000-gal) railcars and ships them to the DSTs via the 204AR unloading facility. As part of operating the facility, small quantities of liquid waste are generated.

A.2.1.9.2 Summary of Activities During March 1991 through February 1992. Following a railcar loading operation, waste transfer lines are flushed to reduce contamination and radiation levels. Each transfer generates approximately 189 L (50 gal) of waste. In the past year, the 340 Facility has made three transfers adding 568 L (150 gal) to the tank waste inventory.

Periodic decontamination activities (i.e., sampling hood, floor sump, and equipment repairs) have resulted in some waste generation. For the past year it is estimated approximately 378.5 L (100 gal) of waste was added to the tank waste inventory.

A.2.1.9.3 Listing of Applicable Documents. None.

A.2.1.9.4 Status of 1992 Activities in Progress. Due to the evaporator shutdown, no large liquid waste generating activities are planned. Once the evaporator is made operational again, the 340 Facility plans to flush out the auxiliary storage tanks to reduce the radiation dose levels. The area is currently categorized as a controlled radiation area with average dose rates exceeding 50 mrem/hr.

A.2.1.9.5 Waste Minimization Activities. Previously, the 340 Facility has flushed both the fill and the vent transfer lines after each railcar loading. The radiation levels and radioactive contamination levels in the vent line have not measurably increased during a transfer operation. The railcar loading procedure was revised in FY 1991 to require a vent line flush only when directed under supervision. When the levels in the vent line exceed the

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safe limit line flushing will be stopped. This has reduced the overall volume of liquid generated from flushing operations at the facility by 50 percent.

A.2.1.9.6 Estimate of Planned Work Activities for 1993. The six 340A auxiliary storage tanks are planned to be flushed of residual solids. It is anticipated that this effort will generate 30.3 to 37.85 m^3 (8,000 to 10,000 gal) of additional waste.

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A.3.0 CURRENT WASTE GENERATORS AT THE 400 AREA

This section documents the studies, activities, and issues which occurred in this area during the period of March 1, 1991, through February 29, 1992.

A.3.1 DESCRIPTION OF FACILITY AND TYPES OF WASTE GENERATED

The 400 Area contains the Fast Flux Test Facility (FFTF), a U.S. Government-owned nuclear reactor specifically designed for the irradiation and testing of nuclear reactor fuels and materials. The FFTF plays a key role in developing and testing fuels and materials for application in fast neutron flux reactors and in testing fusion reactor materials.

This 400-MW fast-breeder reactor is located in a shielded cell in the center of the containment building. The heat generated by the fission process is removed from the reactor by liquid sodium circulating under low pressure through three primary coolant loops. An intermediate heat exchanger in each of these three loops separates the radioactive sodium in the primary system from the nonradioactive sodium in the secondary system. The radioactive primary sodium does not leave the Reactor Containment Building. Three secondary sodium loops transport reactor heat from the intermediate heat exchangers to the air-cooled tubes of the 12 dump heat exchangers.

The FFTF also includes facilities for receiving, conditioning, storing, and installing core components and test assemblies. Examination and packaging capabilities for onsite and offsite shipments and radioactive waste handling are also available at the facility.

A.3.2 GENERATION OF TANK WASTES IN THE 400 AREA

In the 400 Area, radioactive liquid wastes are generated primarily in conjunction with the removal of residual sodium from irradiated reactor components and fuel assemblies in the Interim Examination and Maintenance (IEM) Cell and by the cleaning and decontamination activities conducted in the Maintenance and Storage Facility (MASF). Wastewater, which is generated during the cleaning processes, is stored in a 18.9-m^3 (5,000-gal) tank at the FFTF and in two 18.9-m^3 (5,000-gal) tanks at the MASF. The wastewater is moved from the FFTF to the MASF via an 30.3 m^3 (8,000-gal) railcar and then transferred to the 200 Area Tank Farms via a 75.7-m^3 (20,000-gal) rail tank car. A shipment of the contaminated wastewater to the 200 Area Tank Farms occurs approximately once every two years.

During the past year, 9.8 m^3 (2,600 gal) of wastewater was generated in the IEM Cell and 2,044 L (540 gal) was generated in the MASF. This volume is currently stored in the FFTF and MASF storage tanks. These amounts are consistent with the generation rate over the last several years.

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A.3.3 TANK WASTE MINIMIZATION AT THE FAST FLUX TEST FACILITY AND AT THE MAINTENANCE AND STORAGE FACILITY

The design of the cleaning systems used in the IEM Cell enables the washwater to be recirculated to the greatest extent possible, which minimizes the amount of radioactive tank waste generated by the facility. Current practices generate about 1,892 L (500 gal) of contaminated water with each cleaning episode. The total quantity of wastewater generated each year in the IEM Cell is dependent on the number of reactor assemblies washed.

An annual hydrostatic test is required for the 30.3-m^3 (8,000-gal) tank car which is used to ship waste from the FFTF to the MASF. The testing method includes filling the tank with water. After the test is complete, the water used in the test is shipped to the 200 Area Tank Farms. The amount of washwater generated annually by the IEM Cell and the MASF is less than what is required to perform the test. To further minimize the amount of tank waste generated in the 400 Area, procedures have been upgraded to allow the use of existing wastewater from the two 18.9-m^3 (5,000-gal) tanks at the MASF to help fill the tank car for the required annual hydrostatic test. This results in a substantial reduction in the wastewater volume generated annually.

To further minimize the tank waste generated at the T Plant in the 200 West Area, 36.3 m (9,600 gal) of liquid waste were shipped from the MASF to T Plant for use in hydrostatic testing of a 75.7-m^3 (20,000-gal) tank car. The use of the low-level waste (LLW) from the 400 Area to partially fill the 75.7-m^3 (20,000-gal) tank car reduced the new waste generated at T Plant by 36.3 m^3 (9,600 gal).

A.3.4 FUTURE TANK WASTE GENERATED AS A RESULT OF THE FAST FLUX TEST FACILITY SHUTDOWN OPTION

Since April 1, 1992, FFTF has been on cold standby status; therefore, the future of FFTF and the MASF is undetermined at this time. If the reactor is to be permanently shutdown, the amount of wastewater generated would vary greatly depending upon the method of sodium disposal selected. The possibility exists that up to 1,892.5-m^3 (500,000 gal) of radioactive 50 percent sodium hydroxide waste solution from reacting the liquid sodium drained from FFTF with water will be generated from shutdown activities. This solution will need to be treated as radioactive waste. In addition, 946.3 m^3 (250,000 gal) of slightly contaminated, low-level radioactive rinse water or alcohol could be generated as a result of sodium removal operations in FFTF piping and components after the bulk sodium is drained. If FFTF is to remain on standby or resume operation, the waste generation rate would remain at historic levels.

A.4.0 TANK FARMS

This section documents the studies, activities, and issues which occurred in this area during the period of March 1, 1991 through February 29, 1992.

A.4.1 INTRODUCTION

The tank farms located in the 200 East and 200 West areas of the Hanford Site were built for storing and managing radioactive wastes generated by various production and laboratory operations. The tanks are of two different types; SSTs and DSTs.

A.4.2 DESCRIPTION OF THE FACILITIES

A.4.2.1 Single-Shell Tanks

Between 1943 and 1964, 149 SSTs were built for storing radioactive wastes. These SSTs are located in 12 tank farms, with each tank farm consisting of 4 to 18 SSTs.

The SSTs have volumes of 208 to 3,785 m^3 (55,000 to 1,000,000 gal). One hundred thirty-three of the SSTs are 22.9 m (75 ft) in diameter and 9.1 to 16.5 m (29.75 to 54 ft) high, with nominal capacities of 1,893 to 3,785 m^3 (500,000 to 1,000,000 gal). Sixteen of the SSTs are smaller units of similar design; 6.1 m (20 ft) in diameter and 7.8 m (25.5 ft) high with capacities of 208 m^3 (55,000 gal) each.

The tanks are located below grade with at least 1.9 m (6 ft) of soil covering the tanks to provide shielding and minimize the radiation exposure to tank farm operating personnel. Most of the 1,893- and 2,839-m^3 (500,000- and 750,000-gal) SSTs were built in the form of "cascades" of three or four SSTs each. Waste was transferred to the first SST in the cascade and allowed to overflow into each of the successive SSTs in the cascade through inlet and overflow lines located near the top of the steel liner within in each SST.

Access to each of the SSTs is provided by risers penetrating the domed top of the SSTs. These risers vary in diameter from 10.2 to 106.7 cm (4 to 42 in.). Each of the SSTs have up to 11 risers, with the majority of the SSTs having 3 to 5 risers.

Radioactive waste generated during the various Hanford Site operations was not placed into SSTs after November 1980. While the SSTs are considered to have been taken out of service in November 1980, the 149 tanks continue to hold approximately 140,045 m^3 (37 Mgal) of saltcake, sludge, and interstitial liquid.

A.4.2.2 Double-Shell Tanks

Between 1968 and 1986, 28 DSTs were constructed. Three of these tanks are located in the 200 West Area (241-SY Tank Farm) and 25 tanks are located in the 200 East Area (241-AN, -AP, -AW, -AY, and -AZ Tank Farms). All of these DSTs were constructed at least 5 ft below grade to provide shielding and minimize the radiation exposures to operating personnel. Table A.4-1 provides a chronology of the DST construction.

The four 241-AY and -AZ tanks each have a 3,785 m^3 (1-Mgal) capacity and are designed to store the high-heat-generating neutralized current acid waste (NCAW) from the Plutonium-Uranium Extraction (PUREX) process. These tanks are referred to as aging waste tanks and have airlift circulators for mixing and a vessel ventilation system designed to remove and condense steam.

Table A.4-1. Chronology of the Double-Shell Tank Construction.

Tank farm	Year constructed	Number of tanks	Tank volume m^3 (Mgal)	Comment
241-AY	1968-70	2	3,785 (1.00)	Aging waste tank
241-AZ	1971-77	2	3,785 (1.00)	Aging waste tank
241-SY	1974-76	3	4,315 (1.14)	-
241-AW	1978-80	6	4,315 (1.14)	-
241-AN	1980-81	7	4,315 (1.14)	-
241-AP	1983-86	8	4,315 (1.14)	-

The DSTs use a tank-within-a-tank design to provide double containment for the radioactive liquid and solid wastes. This design ensures that if a leak in the primary shell occurs, the liquid waste will be fully contained within the outer shell.

The freestanding primary tank is about 22.9 m (75 ft) in diameter and 14 m (46 ft) high at the dome crown. The carbon steel in the bottom of the tank ranges from 1.3 to 2.5 cm (0.5 to 1 in.) thick. The primary tank wall thickness ranges from 1.3 to 1.9 cm (1/2 to 3/4 in.) with the dome thickness at 1.0 cm (3/8 in.).

An annular space of 0.76 m (2.5 ft) is provided between the primary tank and the secondary steel tank that allows room for installation of liquid-level and leak detection devices, inspection equipment (such as periscopes), television cameras, photographic cameras, ventilation air supply and exhaust ducts, and equipment for pumping liquid out of the annular space.

Tank dome penetrations in the primary tank and annulus allow for various monitoring and processing activities. Primary tank monitoring activities include measurement of liquid level, sludge level, temperature, and pressure.

A.4.3 DOUBLE-SHELL TANKS OPERATION (MARCH 1991 THROUGH FEBRUARY 1992)

The tank farm facilities at the Hanford Site receive radioactive wastes generated by other Hanford Site waste generators. Tank farm operations are typically characterized as a waste receiver rather than a waste generator. However, in the operation of the tank farms, a variety of additions are made that increase the volume of the wastes in the tanks. These streams are identified because their minimization has the overall effect of reducing the volume requiring treatment for final disposal. Waste from these streams is addressed for the period from March 1991 through February 1992.

1. Saltwell Liquor. The SSTs hold moist solids (salts and sludges) that contain interstitial liquid. Saltwell pumping can remove a portion of the interstitial liquid called saltwell liquor (SWL) from these solids. Through calendar year 1990, 105 SSTs have been interim stabilized, leaving 44 SSTs to be interim stabilized by the end of FY 1995 [Tri-Party Agreement milestone M-05 (Ecology et al. 1990)].

   During the February 1991 to February 1992 time frame 972.094 m^3 (256,828 gal) of pumpable liquid was removed from the SSTs and transferred to DSTs. It is predicted that 15,140 m^3 (4,000,000 gal) will be removed from the SSTs by FY 1995 when the saltwell pumping program is expected to be completed.

2. Airlift Circulator (ALC) Flushes. Salts are periodically flushed from the ALCs in the aging waste DSTs using raw water. The volume of ALC water flushes for the specified time period was 210.6 m^3 (55,651 gal) to aging waste tanks.

3. Aging Waste Ventilation System De-entrainer Flushes. This activity, which is necessary to keep the de-entrainers from plugging, added an undetermined quantity of de-entrainer flush water to the aging waste tanks.

4. Jet Pump Transfers. Waste transferred from catch tanks to DSTs using a jet pump added 25 m^3 (6,602 gal) of motive water to the DSTs.

5. The DST 241-AZ-101 Aging Waste Steam Condensate. The DST 241-AZ-101 contains steam coils to boil water from the aging waste. To prevent these steam coils from freezing during winter weather, a small amount of steam must be allowed through the coils. The aging waste steam coils were not operated during this reporting period and did not add any water to the DSTs.

6. Tank Car Waste Flushing and Water from Recertification. Radioactive waste is shipped by rail tank car to the 200 East Area DSTs from the 100 N, 300, and 400 Areas. The tank car used to transport this waste must be flushed and recertified. The volume of waste generated during these operations was 272 m^3 (71,850 gal).

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7. Flush and Wash. Water is used to periodically wash accumulated solids and salts from measurement equipment. Other equipment must be flushed after use or for maintenance. Equipment wash and flush water and the water added for line flushes after tank to tank transfers were unavailable for this reporting period (March 1, 1992 through February 29, 1992).

8. Evaporator Drainage. No water was transferred added to the DSTs from the 242-A Evaporator Facility during this reporting period. An addition of 201 m^3 (53,075 gal) was made to the DSTs from the 242-S Evaporator Facility during this reporting period (March 1, 1992 through February 29, 1992).

The quantity of new water added to the DSTS during this reporting period, including those water additions cited above and other miscellaneous additions, totalled 1,365 m^3 (360,532 gal).

A.4.4 WASTE MINIMIZATION ACTIVITIES

No waste minimization activities were reported for the period from March 1991 through February 1992.

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A.5.0 EVAPORATORS

This section documents the studies, activities, and issues which occurred in this area during the period of March 1, 1991, through February 29, 1992.

A.5.1 INTRODUCTION

Since the early 1950's, eight evaporator facilities have been used to treat tank wastes at the Hanford Site. The only evaporator facility that is planned for continued operation is the 242-A Evaporator-Crystallizer located in the 200 East Area. The schedule for the 242-A Evaporator-Crystallizer was to remain shutdown during March 1991 through February 1992.

A.5.2 DESCRIPTION OF EVAPORATOR FACILITIES

The evaporator building is divided into rooms housing particular process components or support facilities. The main process rooms are the evaporator room, containing the reboiler and vapor-liquid separator, the condenser room, housing the overhead vapor condensers and condensate collection tank, and the pump room, which contains the slurry pumps. Support rooms include the control room, loading room, heating, ventilation, and air conditioning (HVAC) room, and change rooms.

The 242-A Evaporator is used to reduce the volume of radioactive mixed waste requiring storage in the DSTs. The evaporator uses forced circulation through the reboiler and vapor-liquid separator to heat the waste under vacuum, causing vaporization of water and other volatiles. The vapors from the separator are condensed, retained, and then treated prior to disposal. The slurry product stream is sent back to the DSTs from the evaporator. The volume of the slurry-product stream is significantly less than the volume of the waste feed stream.

A.5.3 TYPES OF WASTE GENERATED

The operation of the 242-A Evaporator-Crystallizer 242-A does not generate new tank waste except when there is a process upset. The following streams are generated:

- Double-shell slurry feed (DSSF), which is returned to DSTs
- Steam condensate from reboiler, which is sent to the 216-B-3 Pond
- Process condensate, which is held for treatment
- Cooling water from the process condenser, which is sent to the 216-B-3 Pond
- Small volume, intermittent wastes such as de-entrainer wash, which are sent to the evaporator pot.

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The slurry returned to the DSTs is not considered an original waste stream for the tank farms.

The small-volume, intermittent wastes such as de-entrainer wash, are sent to the evaporator pot where their identity is lost during evaporation with DSSF.

If there is an upset condition and process condensate becomes contaminated with radionuclides, the process condensate may be returned to a DST. Upset conditions seldom occur and the process condensate is typically not considered a tank waste.

A.5.4 STATUS OF ACTIVITIES IN PROGRESS

Previously, process condensate was discharged untreated to the Hanford Site soil column in the 200 East Area because it was not typically considered a tank waste. This practice has been discontinued and a new collection, treatment, and processing facility is being constructed to treat process condensate.

A.5.5 WASTE MINIMIZATION

An equipment modification was made which replaced the air dryers for the facility process and instrument air. The old equipment used steam to heat the incoming air, and produced a steam condensate waste stream. The new equipment uses electric heaters, thus eliminating this source of steam condensate which previously exited into the 216-B-3 Pond System.

A.5.6 PLANNED WORK

The 242-A Evaporator is expected to resume operations in 1993. Projected volume reductions for the first waste reduction campaign is approximately 9,463 m^3 (2,500,000 gal).

A.6.0 PLUTONIUM FINISHING PLANT

This section documents the studies, activities, and issues which occurred in this area during the period from March 1, 1991, through February 29, 1992.

A.6.1 INTRODUCTION

The Plutonium Finishing Plant (PFP) is located in the 200 West Area of the Hanford Site. The PFP has the primary mission of plutonium processing, handling, and storage. Stabilization of plutonium scrap to plutonium oxide, waste treatment, product storage, and packaging for shipment are the principal operations conducted at the PFP. Plutonium metal will not be produced at the PFP because of changes in the defense production mission at the Hanford Site.

A.6.2 RECAP OF MARCH 1, 1991 THROUGH FEBRUARY 29, 1992 ACTIVITIES

A.6.2.1 Planned Treatment of Plutonium Finishing Plant Waste

Present plans are to develop and utilize a PFP Waste Solidification Process (Project C-130) where the process waste will be treated for the removal of organics, nitrates, and water, and then solidified. The resultant solids will either contain transuranic (TRU) or low level amounts of TRUs which will be solidified into 208 L (55-gal) drums and certified for final emplacement at the WIPP site in Carlsbad, New Mexico, or for burial at the Hanford low level burial site. Project C-130 was planned as a FY 1995 line item, which means that the design for the PFP Waste Solidification Process was scheduled to start in FY 1995. Funding for the project was not provided in the FY 1992 budget. The project has not been canceled but has been placed on hold until funding is allocated.

A.6.2.2 Plutonium Reclamation Facility Process Modification

Bypassing of the Outside Air (OA) Column during plutonium-only and uranium depletion operations, as described in the 1990 *Annual Report of Tank Waste Treatability*, will take place when the Plutonium Reclamation Facility (PRP) starts up. The PRF is scheduled for restart in the latter part of calendar year (CY) 1992.

A.6.2.3 Project C-031H

Project C-031H, the PFP Liquid Effluent Treatment Facility Upgrade, consists of removal and replacement of four of the five waste tanks in the 241-Z Building. Accordingly, the concrete tank vaults containing these tanks will be repaired. Each vault will then be lined with stainless steel. Redundant tank level measuring devices will be installed on the new tanks.

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Four new encased and monitored transfer lines from the plant to the 241-Z Building will also be installed. The existing transfer lines will not be removed but will be left in place.

The waste tanks are used for storage and treatment of transuranic aqueous wastes from the PRF, the Remote Mechanical C-Line (RMC) and the development and analytical laboratories. After treatment, the wastes are transferred to tank farms. The tanks will be replaced two at a time, allowing the remaining three tanks to store and treat wastes generated by the plant during the construction. Project completion is scheduled for December, 1995. The fifth tank, D5, will not be removed but will be taken out of service and left in place.

A.6.3 WASTE GENERATED AND CURRENT INVENTORY

Approximately 5.7 m^3 (1,500 gal) of liquid wastes were generated in CY 1991. No treatment chemicals were added to the waste tanks because no transfers were made to the 224-X Tank Farm during this reporting period. Approximately 31.4 m^3 (8,300 gal) of water were added to the four 241-Z waste tanks from a water leak. In summary, there were approximately 47.65 m^3 (12,590 gal) of liquid waste in the D-4, 5, 7, and 8 waste tanks on December 31, 1991. Approximately 16.24 m^3 (4,290 gal) of the above total were wastes generated in the PFP.

A.6.4 WASTE MINIMIZATION ACTIVITIES

A.6.4.1 Plutonium Reclamation Facility Process Modification

In addition to the modifications previously described in the *1990 Annual Report of Tank Waste Treatability*, the following modifications for the abatement of CCl_4 emissions are being investigated.

- During the PFP startup, a water cap will be in place between the CCl_4 and the air pulser on all pulse extraction columns to minimize CCl_4 emissions, because the extractions columns are known to leak.
- Investigations are continuing to find a suitable replacement for the CCl_4 solvent that is more environmentally acceptable.

A.6.4.2 PFP Waste Minimization

Waste minimization activities described in the 1990 and 1991 reports are continuing. Additional activities include the following.

- Twenty-three 208-L (55-gal) drums of 45 percent KOH purchased for use in the RMC hydrogen fluoride scrubber system have been designated as surplus material, because HF gas will not be used in the RMC process. The surplus KOH will be used for hydroxide ion adjustment in the D-5 waste treatment tank in place of the normally-used NaOH until all of the KOH is consumed.

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- An aqueous ferric nitrate solution currently is used in the D-5 waste treatment tank for the formation of the solids required for tank farm transfers. There are approximately 136 kg (300 lbs) of solid ferric nitrate in storage at PFP that was previously used for makeup of the ferric nitrate solutions. A procedure is being updated that will allow the use of the stored, solid ferric nitrate to be dissolved in water and used as the makeup solution to supplant the ferric nitrate solution presently used.

- The volume of aqueous effluent samples from the PFP crib has been reduced from 1 L to 1/2 L. The sample size reduction resulted in a decrease of approximately 1,900 L (500 gal) of waste liquid that would have been in the 222-S Laboratory waste tanks.

A PFP staff member has developed a pollution prevention plan which he has presented to Westinghouse Hanford Company (Westinghouse Hanford) employees at the FFTF, PFP, and Grout Treatment Facility (GTF). At the conclusion of each presentation, attendee participation was solicited. Suggestions and ideas concerning pollution prevention and/or waste minimization were discussed. Ideas generated at the PFP presentation were tabulated and evaluated by the PFP pollution prevention team for general applicability to PFP and other Hanford Site facilities.

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A.7.0 PUREX PLANT

This section documents the studies, activities, and issues which occurred in this area during the period of March 1, 1991, through February 29, 1992.

A.7.1 INTRODUCTION

The PUREX Plant was designed to reprocess irradiated nuclear reactor fuels for the recovery of uranium and plutonium. The last fuel reprocessing run (the stabilization run) was completed in March 1990. Since October 1990 the PUREX Plant has been in cold standby mode.

A.7.2 DESCRIPTION

A.7.2.1 Facility

The PUREX Plant is located in the southeast corner of the 200 East Area of the Hanford Site. The PUREX Plant comprises several buildings and support facilities.

The primary structure is the 202A Building. The 202A Building is a reinforced concrete canyon structure 304 m (1,000 ft)-long, 36.3 m (119 ft) -wide (at its maximum width) and 30.4 m (100 ft)-high, with approximately 12.2 m (40 ft) of this height below grade. It contains a "canyon" with processing cells, a laboratory, various support systems and galleries, and administrative offices.

Several other buildings associated with the PUREX Plant complex include the following; several mobile office trailers, structures associated with various support functions, two long storage tunnels, two small tank farms, warehouses, and several materials storage areas.

A.7.2.2 PUREX Process

The PUREX process and associated equipment were designed to chemically extract plutonium and uranium from irradiated metal nuclear reactor fuel. Because of the radioactive materials being reprocessed, the system has been designed for remote operation and maintenance. The reprocessing equipment is located in the process cells within the PUREX canyon. The PUREX Plant is currently configured to reprocess zircaloy clad fuel from N Reactor.

Plutonium and uranium separations begins with the batch dissolution of the fuel cladding followed by batch dissolution of the spent reactor fuel itself. The dissolved fuel constituents are then fed into a continuous aqueous/organic solvent extraction process system. The solvent extraction process system separates mixed fission products from the plutonium and uranium. The plutonium and uranium are then separated from each other and purified in subsequent reprocessing operations. The final products are uranyl nitrate hexahydrate (UNH) and either plutonium oxide or plutonium nitrate.

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A.7.2.3 Waste Types

The wastes produced by the PUREX Plant fall into four general types: neutralized current acid waste (NCAW), neutralized cladding removal waste (NCRW), miscellaneous wastes, and solvent recovery wastes. The NCAW is the aqueous high-salt waste from the first-cycle solvent extraction column in the solvent extraction process system. The NCAW is also referred to as neutralized zirflex acid waste (NZAW). The NCRW results from the dissolution and subsequent removal of the zircaloy cladding from the spent N Reactor fuel by means of the zirflex batch dissolution process. The miscellaneous wastes come from various sources throughout the plant. The solvent recovery wastes result from washing and regenerating the non-regulated organic solvent (tributyl phosphate/normal paraffin hydrocarbon) used in the PUREX solvent extraction process.

The NCAW, NCRW, and the miscellaneous waste are all radioactive mixed wastes regulated by the U.S. Environmental Protection Agency (EPA) and the Washington State Department of Ecology (Ecology). The solvent recovery wastes are radioactive wastes controlled administratively by the U.S. Department of Energy (DOE). The pH of all wastes is adjusted to a value greater than 12. Sodium nitrite is then added to the waste solution for purposes of corrosion control prior to transfer to the DSTs for interim underground storage.

During transition-to-standby and cold standby, the principal type of waste being generated is miscellaneous waste. A small amount of solvent recovery waste may also be produced. The NCAW is only generated during fuel reprocessing and will not be generated during cold standby. The single batch of NCAW generated between March 1, 1991 through February 29, 1992, was associated with transition-to-standby equipment flushing operations rather than fuel reprocessing.

A.7.3 RECAP OF ACTIVITIES FROM MARCH 1991 THROUGH FEBRUARY 1992

In October 1990, the U.S. Department of Energy, Richland Field Office (RL) put the PUREX Plant on cold standby. Cold standby may be defined as placing the plant into a safe and environmentally sound condition that does not compromise future fuel reprocessing capability.

The plant has been in a transition-to-standby mode of operation for this entire reporting period. Most of the requirements for physical modifications to meet the cold standby status have been completed. The plant activities have included equipment maintenance, isolation of water, steam, and chemical lines, and general surveillance. There are several administrative issues, primarily related to the Operational Safety Report (OSR) documentation, that have not yet been resolved. These issues will require resolution before the plant can enter the standby condition. The PUREX Plant will remain in either the transition-to-standby condition or standby condition until additional guidance on the plant status and future activities is provided by RL and/or U.S. Department of Energy-Headquarters (HQ).

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A.7.4 LISTING OF APPLICABLE DOCUMENTS

No studies on tank waste minimization were published between March 1, 1991, and February 29, 1992.

A.7.5 STATUS OF CY 1991 ACTIVITIES IN PROGRESS

As part of the transition-to-standby activities, various plant systems are being isolated from the steam and water supplies to reduce waste generation. System isolation is continuing. The waste volume being saved is not readily quantifiable at the present time.

PUREX Plant Uranium Storage Tank Farm (203-A Area) equipment is being modified to divert steam condensate and rainwater from the DSTs to the soil column via the PUREX chemical sewer effluent stream. The permanent modifications have not been completed yet. During CY 1991, about 570 m^3 of steam condensate was diverted to the ponds instead of the DSTs.

A.7.6 CURRENT INVENTORY AND AMOUNTS GENERATED

A.7.6.1 Tank Waste Inventory

None of the tanks used to accumulate tank waste in the PUREX Plant are permitted for further storage. The tanks used to collect the NCAW, NCRW, and miscellaneous waste are permitted as 90-days accumulation tanks and do not store tank waste. The solvent recovery tanks contain radioactive nonregulated waste and do not require permitting. As a matter of operating practice, solvent recovery wastes are also transferred to tank farms within 90 days.

A.7.6.2 Tank Waste Generated

Between March 1, 1991, and February 29, 1992, the following types and amounts of tank wastes were transferred from the PUREX facility to the tank farms DSTs:

- NZAW waste: 20 m^3 (5,279 gal)
- NCRW waste: 0 m^3
- Miscellaneous waste: 285 m^3 (75,300 gal)
- Solvent recovery waste: 0 m^3.

A.7.7 WASTE MINIMIZATION ACTIVITIES

A broken water main increased the contamination levels in the stack plenum. Use of a vacuum cleaning system and squeegees for contamination reduction avoided the generation of about 16 m^3 of contaminated liquid when compared to past cleaning efforts.

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A.7.8 ESTIMATE OF PLANNED WORK ACTIVITIES FROM MARCH 1991 TO FEBRUARY 1992

A major expected effort involving tank waste are the process waste assessments to meet both DOE-HQ and Ecology requirements for identifying waste reduction opportunities. Work on the process waste assessment for the PUREX Plant tank wastes has been on hold pending negotiations between DOE-RL and Ecology on the application of the *Washington Administrative Code* (WAC) 173-306 *Pollution Prevention Plans* to the Hanford Site. As of February 1992, these negotiations are still in progress. The final details, scope, and schedule will not be established until the negotiations are completed.

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A.8.0 B PLANT

This section documents the studies, activities, and issues which occurred in this area during the period of March 1, 1991, through February 29, 1992.

A.8.1 DESCRIPTION OF FACILITY

B Plant is designed to remotely process radioactive materials with minimal radiation exposure to operators. The first mission of B Plant was to reprocess spent fuel between 1945 and 1952 using the bismuth phosphate process. B Plant was refurbished for Mission 2 (1965 to 1985) to recover and purify cesium and strontium from newly generated current acid waste (CAW) and from stored wastes in tanks (NCAW). The B Plant canyon as well as other major areas of the facility have initiated general cleanup activities.

A.8.2 STATUS OF CURRENT ACTIVITIES

A.8.2.1 Support to the Waste Encapsulation and Storage Facility for Storage of Cesium and Strontium Capsules

B Plant currently provides demineralized water to the Waste Encapsulation and Storage Facility (WESF) for pool-cell storage of cesium and strontium capsules. B Plant also provides treatment for low-level radioactive liquid waste produced at WESF, as well as lag storage for radioactive solid waste generated at WESF.

A.8.2.2 Management of an Existing Inventory of Radioactive Liquid Waste

Radioactive liquid waste is currently in storage at B Plant. This waste includes organic solutions containing cesium and strontium as well as some organic solvents. These liquid wastes exist at B Plant as a result of previous missions. Several tanks at B Plant currently contain NCAW waste, which was transferred to B Plant for the purpose of waste pretreatment studies. Plans are currently being developed to remove the liquid inventory from B Plant.

A.8.2.3 Management of an Existing Inventory of Radioactive Solid Waste

B Plant currently stores drums of radioactive solid waste in cell 4. These drums of waste, as well as several waste piles (used jumpers and miscellaneous piping) stored on the canyon deck, are the result of both past and current operations at B Plant and WESF.

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There are currently several megacuries (MCi) of radioactively contaminated materials in B Plant. Buried HEPA filters, the process equipment, and the structure itself are the major sources of radiation. Strontium-90, and ^{137}Cs, deposited during Mission 2, are the principal radionuclides contributing to the radiation dose levels in B Plant.

A.8.2.4 Treatment of Low-Level Waste Generated by Operation of Plant Ventilation Systems

The pH of the process condensate is chemically adjusted for low-level radioactive liquid wastes generated in B Plant and WESF, before transfer to the DSTs.

A.8.2.5 Process Condensate Treatment Facility

A study is currently underway to evaluate the options for treatment and discharge of process condensate which is generated by the operation of the B Plant concentrator. The results of the 240 BAT studies also will be incorporated into this effort.

A.8.3 WASTE MINIMIZATION ACTIVITIES

Several waste minimization activities have been initiated at B Plant during this reporting period. The following activities are directly related to the overall DST waste minimization effort.

A.8.3.1 Suspend Tank Farm Flushes

Past operations procedures at B Plant provided for flushing the transfer line to tank farms after each waste transfer to prevent solids buildup in the transfer line. This procedure added about 14.2 m^3 (3,750 gal) of supplemental waste to each transfer of waste to the DSTs. Current procedures call for suspension of flushing prior to the receipt of solids testing results and to flush only when the solids content of the waste exceeds 4 percent. This practice, implemented in 1990, has reduced the volume of waste transferred to the DSTs by approximately 567.8 m^3 (150,000 gal) in this reporting period (March 1, 1991 through February 29, 1992).

A.8.3.2 Minimize Tank Liquid Heel Replacement

Tank liquid heels, also known as water seals, have been maintained with demineralized water according to previous operating procedures at B Plant. These water seals were used to prevent contamination between tanks connected to a common ventilation system. This practice was discontinued in June 1990. The maintenance of tank liquid heels is now accomplished with low-level radioactive liquid waste. A waste reduction of about 151.4 m^3 (40,000 gal) was affected during this 12-month reporting period.

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A.8.3.3 Rerouting of Waste and Elimination of Steam Jet Dilution

Low-level liquid waste has been rerouted through tanks equipped with water pumps rather than using steam jets; i.e., tank 24-1 to tank 25-1 vs. tank 24-1 to tank 23-3 to tank 23-1 to tank 25-1. This practice has eliminated the need for steam jetting, which, in turn, has eliminated a source of liquid dilution. This practice has resulted in a waste reduction of approximately 64.3 m^3 (17,000 gal) during this 12-month reporting period.

A.8.4 CURRENT INVENTORY AND/OR AMOUNTS GENERATED

During the reporting period from March 1, 1991 through February 29, 1992, B Plant transferred 1,003 m^3 (265,000 gal) of low-level radioactive waste to the DSTs. This waste consists primarily of steam condensate which is generated by operation of essential plant ventilation systems.

A.8.5 ESTIMATE OF PLANNED WORK ACTIVITIES

Three primary activities currently are planned for B Plant as follows.

- Preparation for future missions will be initiated by cleanout and stabilization of the B Plant canyon and hot-cells.
- Operation of the LLW concentrator will provide system optimization and characterization of the B Plant process condensate and steam condensate effluent streams.
- Solid waste volume reduction will be implemented by use of a jumper cutter.

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A.9.0 222-S LABORATORY COMPLEX

This section documents the studies, activities, and issues which occurred in this area during the period of March 1, 1991, through February 29, 1992.

A.9.1 DESCRIPTION OF LABORATORY-COMPLEX FUNCTION, FACILITIES, AND WASTE

A.9.1.1 Laboratory-Complex Function

The 222-S Laboratory Complex (222-S Complex), in the southeast corner of the 200 West Area, consists of the 222-S Laboratory (222-S), the 222-SA Standards Laboratory, and several ancillary facilities. The main facility of the complex consists of the 222-S Laboratory, which provides analytical chemistry and radiological services.

The current mission of the 222-S Complex is to provide quality analytical services supporting the Hanford Site processing units with current emphasis on waste management, chemical processing, and environmental functions for the following facilities:

- B Plant
- U Plant
- Tank farms
- 242-A and 242-S Evaporators
- GTF
- WESF
- PUREX
- PFP.

Quality analytical services are also provided in support of general process development/upset activities.

Currently the 222-S Complex is being upgraded to support *Resource Conservation and Recovery Act of 1976* analytical protocols and programs for environmental restoration and DST characterization activities for the Hanford Site.

A.9.1.2 Facilities

The 222-S Laboratory is a two-story, above-ground building, 98-m (322-ft) long and 32.6-m (107-ft) wide. This structure is divided into laboratory support spaces, offices, a multi-curie wing, and supplemental service areas. It has facilities for waste disposal and decontamination, and systems for ventilation, radiation monitoring, and fire protection, including alarms.

The first floor of 222-S is divided into three general sections; west, east, and central. The west section contains a lunchroom, offices, and changerooms. This section is kept free of radioactivity and toxic chemicals. The central section has service areas and laboratories where toxic chemicals and low-level radioactive materials are analyzed, and intermediate-level

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radioactive samples are also analyzed occasionally. The east section, commonly known as the multi-curie section, contains laboratories and cells in which intermediate-level radioactive materials are analyzed.

The 219-S Waste Handling Facility (219-S) has three storage tanks in which liquid acid waste from 222-S can be received, stored temporarily, and neutralized. From this facility, neutralized waste, which may contain radionuclides, is transferred to the tank farms. A 2.65 m^3 (700-gal) sodium-hydroxide supply tank is also located in this facility.

A.9.1.3 Waste

Most waste generated at the 222-S Complex derives from analytical activities in 222-S. Waste acid from 222-S is pumped to the 219-S Waste Handling Facility. There are three tanks in 219-S (TK-101, TK-102, and TK-103) that receive hazardous and radioactive liquid waste. Waste acid solution from 222-S is pumped to either TK-101 or TK-103. From these tanks, the waste is transferred to TK-102 for pH adjustment using sodium hydroxide. As needed, sodium nitrite is added to the solution, which raises its nitrite concentration to levels meeting tank farm specifications. Then to ensure adequate mixing of the waste constituents, the solution is agitated. After these steps are completed, the neutralized acid waste is ready for transfer to the tank farms for long-term storage until it can be disposed of permanently.

The types and respective concentrations of wastes typically resulting from laboratory activities are shown in Table A.9-1. Figure A.9-1 illustrates typical concentrations of 222-S waste. The volumes of waste generated, chemical compositions, radionuclide constituents and concentrations, and amounts of solids may vary depending on the analytical activities used to support different programs.

Intermediate-level radioactive waste streams are pumped to tank-101 of 219-S. These streams originate from hood drains, decontamination hood No. 16, hot laboratory sinks, and inductively coupled plasma analyzers.

High-level radioactive waste streams are pumped to tank-103. These streams originate from hot-cell drains, jet-suction vacuum (slurping) operations performed at decontamination hood no. 16, the 1-F manipulator-repair hood drain, the atomic-absorption spectrophotometer hood drain, and from the hot tunnel sumps.

A.9.2 WASTE MINIMIZATION

Waste minimization plans affecting the 219-S tanks are currently being investigated to help reduce the amount of liquids being disposed of to the tanks. Two examples of waste minimization activities currently being considered are:

- Reducing the amount (volume) of sample being sent by the generator. This procedure would minimize the quantity of sample waste because the total delivered sample volume is not always used in the laboratory analysis.

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Table A.9-1. 222-S Laboratory Waste Composition.

Chemical	Composition
Liquids	
Carbonate	5.0 E-03 M
Total organic carbon	1.0 E+00 g/L
Fluoride	1.0 E-03 M
Nitrite	2.5 E-02 M
Nitrate	1.0 E-01 M
Phosphate	5.0 E-03 M
Sulfate	2.0 E-02 M
Sodium	2.5 E-01 M
Hydroxide	1.0 E-01 M
Radionuclides	
Total alpha	5.0 E-06 Ci/L
Total beta	2.0 E-04 Ci/L
^{137}Cs	5.0 E-05 Ci/L
$^{89,90}Sr$	3.0 E-05 Ci/L
Plutonium	4.0 E-05 g/L
Uranium	1.0 E-02 g/L
Solids	
Percent	0.00 E+0

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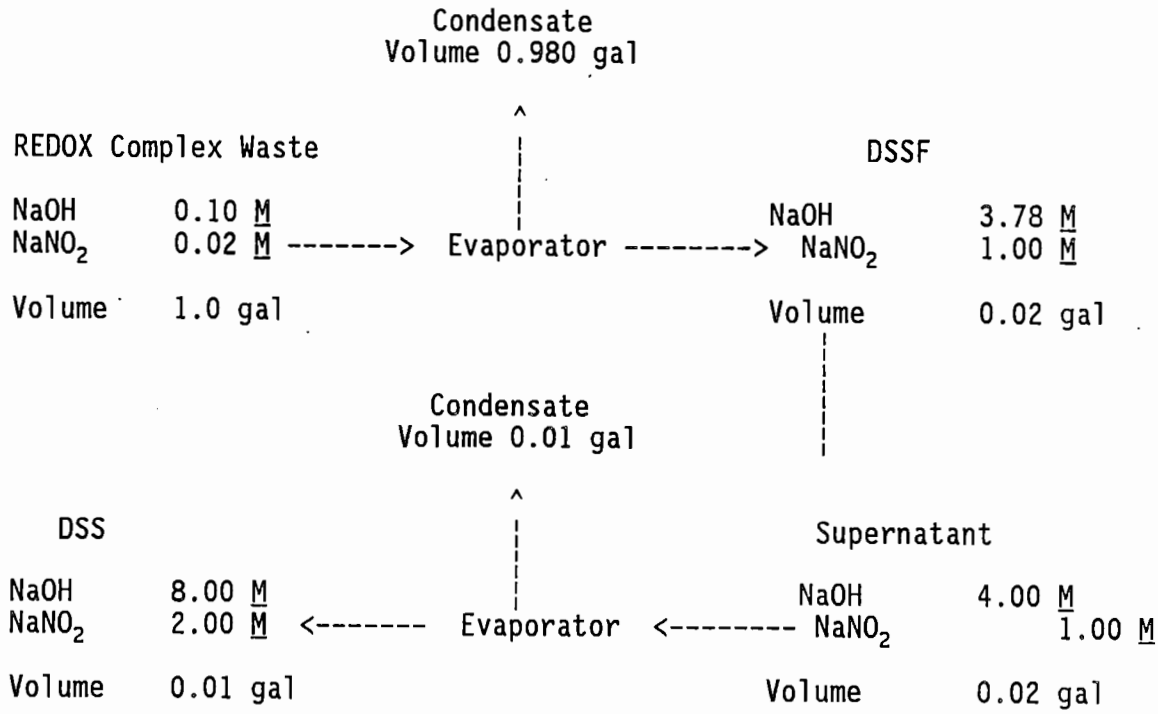
Figure A.9-1. Concentration of 222-S Laboratory Waste.



- Returning unused sample portions to the generator of the sample for disposal, which would result in a reduction of aqueous sample volumes being dumped to the waste tanks.

A.9.3 STATUS OF ACTIVITIES IN PROGRESS

The projected volumes of waste are based on facility operating plans, target waste-generation rates, and the SST and DST characterization schedules.

From FY 1992 through FY 1994, ten SST and DST core samples per year are scheduled for analysis. This schedule increases to 20 core samples per year from FY 1995 through FY 2015. These projections will be adjusted if current schedules change. Extensive chemical and radionuclide analysis also will continue through FY 1992, with subsequent projections being based on the results of the preceding analytical data.

During the twelve month period from March 1, 1991 through February 29, 1992, 110.9 m^3 (29,294 gal) of liquid waste was transferred to tank 204 AR in the 200 East Area Tank Farms.

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A.10.0 T PLANT

This section documents the studies, activities, and issues which occurred in this area during the period of March 1, 1991, through February 29, 1992.

A.10.1 FACILITY DESCRIPTION

T Plant is located in the 200 West Area of the Hanford Site. The T Plant's primary mission is equipment decontamination and refurbishment. The head end of the 221-T canyon building houses the Containment Systems Test Facility. This facility was used to perform experimental testing which requires containment or isolation. The T Plant waste system handles radioactive liquid waste from decontamination activities in the hot-cells, the railroad tunnel, the 2706-T Building, and the head end. The railroad tunnel generates waste from decontaminating railroad cars and multipurpose transfer boxes.

Most of the waste from cells in T Plant consists of water with settled solids generated during decontamination activities. Each cell in the 221-T Canyon has a 15-cm-dia. drain line that allows wastewater to drain into the canyon's 61-cm-dia. sewer line. Potentially contaminated wastes from the head end are also drained through a 15-cm line into the canyon's 61-cm-dia. sewer line. This line empties into tank 5-7 in the canyon. The waste in tank 5-7 is transferred to tank 15-1. In tank 15-1, the waste is sampled, analyzed, then sent to tank farms via the cross-site transfer line or by certified railcar. If the waste is to be delivered via the cross-site transfer line, it is chemically treated to meet tank farms' storage specifications prior to the transfer operation.

A.10.2 SUMMARY OF MARCH 1991 THROUGH FEBRUARY 1992 ACTIVITIES AND WASTE GENERATED

During this time period, T Plant was under limited operational status and generated only 311.6 m^3 (83,082 gal) of waste. The majority of this waste was generated from the addition of water to the rail cars for purposes of railcar certification. The composition of this waste is presented in Table A.10-1. The radioactivity levels of this waste is given in Table A.10-2 for the most significant radionuclides. These data, obtained from process sample data, represent an arithmetic average of the laboratory analysis results. Since April 3, 1991 protocol samples also have been taken, but no analytical data has been made available during this reporting period.

A.10.3 STATUS OF ACTIVITIES IN PROGRESS

T Plant decontamination operations are still in a limited operational mode while planned facility upgrades are being completed and operating procedures are being updated and revised.

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Table A.10-1. T Plant Waste Chemical Characteristics.

Chemicals	Composition
PO_4	2.48×10^{-3} M
NO_2	6.42×10^{-4} M
NO_3	2.54×10^{2} ppm
Pb	7.15×10^{-1} ppm
Ag	1.05 ppm
Cd	0.007 ppm
Ba	0.795 ppm
As	0.1 ppm
Hg	27 ppm
Se	0.12 ppm
Cr	1.01 ppm
pH	9.4
Specific gravity	0.994
Percent solids	6.87
Separable organics	None

Table A.10-2. T Plant Waste Radiological Characteristics.

Radionuclides	Concentration
^{238}Pu	7.83×10^{-9} Ci/L
$^{239/240}Pu$	9.33×10^{-8} Ci/L
^{241}Pu	1.50×10^{-6} Ci/L
^{242}Pu	1.48×10^{-11} Ci/L
^{234}U	1.92×10^{-8} Ci/L
^{235}U	4.28×10^{-4} g/L
^{236}U	1.14×10^{-5} g/L
^{238}U	6.03×10^{-2} g/L
^{137}Cs	2.82×10^{-6} Ci/L
^{154}Eu	1.79×10^{-6} Ci/L
^{152}Eu	1.05×10^{-6} Ci/L
^{155}Eu	4.33×10^{-6} Ci/L
^{60}Co	1.22×10^{-7} Ci/L
$^{89/90}Sr$	8.35×10^{-6} Ci/L
^{241}Am	9.51×10^{-8} Ci/L
Total alpha	2.31×10^{-7} Ci/L
Total beta	1.86×10^{-5} Ci/L

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A.10.4 CURRENT INVENTORY AND/OR AMOUNTS GENERATED

The current tank waste inventory is 50.1 m^3 (13,233 gal). Until decontamination operations are resumed, the waste volumes produced will be limited.

A.10.5 WASTE MINIMIZATION ACTIVITIES

The use of plastic and paper for contamination control during work activities within the tunnel has resulted in a reduction in the requirements for post-job decontamination. This, in turn, has reduced the total amount of waste generated.

Liquid LLW generated by T Plant also is used for hydrotesting of railcars, which reduces the amount of water that must be added to the railcar for these tests.

A.10.6 ESTIMATE OF PLANNED WORK ACTIVITIES FOR FISCAL YEAR 1993

As previously stated, T Plant decontamination operations have been limited during FY 1992. The following activities are planned for FY 1993:

- Start construction of a hard pipe transfer line from tank 15-1 to access a railcar
- Complete the Readiness Review and resume operations at 2706-T
- Conduct a Canyon Operations Readiness Review.

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A.11.0 HANFORD WASTE VITRIFICATION PLANT

The Hanford Waste Vitrification Plant (HWVP) currently is scheduled for start up in 1999. The low-level waste generated at this facility will be returned to the DST farms for storage treatment and for disposal as grout waste.

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A.12.0 GROUT TREATMENT FACILITY

This section documents the studies, activities, and issues which occurred in this area during the period of March 1, 1991, through February 29, 1992.

A.12.1 DESCRIPTION OF FACILITY AND TYPES OF WASTE GENERATED

A.12.1.1 Description of Facility

The GTF, located in the 200 East Area of the Hanford Site, has the primary mission of permanently disposing of LLW. These LLWs will be blended with cementitious materials for immobilization and solidification in below-ground vaults. The GTF includes the Dry Materials Facility (DMF), the Grout Processing Facility (GPF), and the Grout Disposal Facility (GDF).

The DMF has the primary purpose of receiving, storing, and blending the dry cementitious grout materials. Materials used in this facility include portland cement, fly ash, and blast furnace slag. No radioactive materials are handled at the DMF.

The GPF has the main purpose of receiving radioactive liquid LLW from the 241-AP Tank Farm feed tank, mixing it with the dry-blend materials from the DMF, and transferring the resultant grout mixture to a disposal vault.

The GDF is where the grout disposal vaults are located. The grout slurry mixture is pumped into the vault and cures into a hardened grout product. Liquid waste generated by the grout process or excess water and leachate liquid from the vault during the setting and curing process is returned to the tank farms for processing. Flush liquid results in additional liquid waste to be recycled.

A.12.1.2 Type of Waste Generated

The GTF has generated mixed, low-level radioactive and chemically hazardous liquid waste [approximately 196.8 m^3 (52,000 gal) in the last 2 years].

A.12.2 WASTE MINIMIZATION ACTIVITIES

The waste minimization plan has the primary purpose to reduce the volume, weight, or toxicity of all regulated waste generated at the GTF to the extent practical. Areas addressed in the plan include; organizational responsibilities, employee training, employee participation and incentive programs, and incorporation of waste minimization as part of the design process for new projects or designs.

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A.12.2.1 Employee Training

As part of general training for new employees, waste minimization training is included. General waste minimization training is provided to all employees of the GTF via waste minimization team awareness presentations and for hazardous waste shippers as part of the Hazardous Waste Shipment Certification training. Specific training and application of waste minimization techniques will be provided on an individual or group basis, as appropriate, by the respective manager or supervisor. The manager or supervisor is responsible for establishing employee responsibilities, assignments, and goals. Each group will keep a record of waste minimization training.

A.12.2.2 Employee Participation and Incentive Program

An employee participation and incentive program is part of the waste minimization plan at the GTF. Promotion and application of employee incentives appear to be a good way to minimize waste generation and to maximize the use of good operating procedures. The incentive program has several components.

- Encourage employees to submit suggestions as Productivity Improvement and Cost Effectiveness Program (PRICE) proposals or Great Ideas.
- Encourage employees to submit suggestions to the Westinghouse Hanford waste minimization specific incentive program (currently being developed).
- Encourage employees to submit on-the-job waste minimization ideas directly to the GTF Waste Minimization Team with certificates and other rewards for this program.

A.12.2.3 New Projects and Designs

New projects and designs will be required to include waste minimization as an integral part of the design process. To accomplish this, the GTF waste minimization representative will review any proposed new construction and major grout process changes to ensure that waste minimization has been considered. New construction presently includes four grout disposal vaults and modification to tank 241-AP-104 for use as a second feed tank. New construction under consideration is a Grout Failed Equipment Handling Facility to stage contaminated failed equipment.

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WAC 173-306, "Pollution Prevention Plans," *Washington Administrative Code*, as amended.

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