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West Valley Demonstration Project

Preconceptual Design Study for Solidifying High-Level Waste

Compiled by O. F. Hill
Project Manager, J. R. Carrell

April 1981

Prepared for the U.S. Department of Energy
under Contract DE-AC06-76RLO 1830

Pacific Northwest Laboratory
Operated for the U.S. Department of Energy
by Battelle Memorial Institute

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West Valley Demonstration Project

PRECONCEPTUAL DESIGN STUDY FOR SOLIDIFYING HIGH-LEVEL WASTE

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FOREWORD

On October 1, 1980, President Carter signed Public Law 96-368, known as the West Valley Demonstration Project Act. This Act directs the Department of Energy to carry out "a high-level radioactive waste management demonstration project at the Western New York Nuclear Service Center in West Valley, New York, for the purpose of demonstrating solidification techniques which can be used for preparing high-level radioactive waste for disposal." The New York Nuclear Service Center is the site of the chemical reprocessing plant that was operated by Nuclear Fuel Services, Inc., to process spent nuclear fuels. The Department of Energy assigned the Pacific Northwest Laboratory, which is operated by Battelle Memorial Institute, the lead in conducting a preconceptual design for a solidification process to solidify the waste. This work is being performed in two parts: 1) a preconceptual design for a solidification process for the high-level waste and 2) a broader preconceptual design study that includes all operations necessary to retrieve and solidify the liquid waste, to transport all waste to a federal repository (high-level and transuranic waste) or to a licensed burial ground (low-level waste), and to decontaminate and decommission the tanks, equipment and other facilities in which the high-level waste was stored or processed. In addition to these two design studies, a separate study that assesses several alternative solidification processes and waste forms has been prepared.

The preconceptual design study for the solidification process is the subject of this report. The purpose of this study is to provide a basis for determining the feasibility of solidifying the high-level wastes in the existing facilities at the Western New York Nuclear Service Center. Because of this latter restraint, the study was done in more depth than is usually required for a preconceptual design. Some preliminary design details were necessary.

This study also provides an engineering basis for the solidification portion of the broader preconceptual design for the West Valley Demonstration Project. Vitro Engineering Corporation in Richland, Washington, provided their architectural and engineering services to the Pacific Northwest Laboratory for the preconceptual design for the chemical treatment and solidification processes.

The purpose of the broader preconceptual design is to provide the bases for the conceptual design and preliminary program cost estimates. An environmental impact statement is being prepared by the Argonne National Laboratory. The environmental impact statement and the broader preconceptual design report will provide supportive background for the preparation of a safety analysis report for the project.

The preconceptual design study for the solidification process discussed in this report identifies ancillary processes and steps required to make the West Valley Demonstration complete, but it does not provide the technology. These are treated in the broader report.

We are indebted to prior studies that have been performed relating to the handling of the high-level wastes at West Valley. These include, but are not limited to Alternative Processes for Managing Existing Commercial High-Level Radioactive Wastes (Nuclear Regulatory Commission 1976); Western New York Nuclear Service Center Companion Report (Department of Energy 1978); Report to the President by the Interagency Review Group on Nuclear Waste Management (Department of Energy 1979); and Report of the New York State Energy Research and Development Authority with Respect to Federal Acquisition of the West Valley Facility (New York State Research and Development Authority 1977). We further acknowledge the considerable effort and contributions of the staff of the Vitro Engineering Corporation, Richland, Washington, in the development of the design data and cost estimates. We greatly appreciate the skillful and helpful assistance and suggestions provided by our technical editor, Susan A. McCullough. Finally, we note that the study will undergo an independent cost estimate review by the Los Alamos Technical Associates, in which they will evaluate the validity of the estimates and will make recommendations for adjustments and contingency factors. This review will be reported separately.

JR Carrell
J. R. Carrell
Project Manager

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1.0 SUMMARY

This report presents a preconceptual design study for processing radioactive high-level liquid waste presently stored in underground tanks at the Western New York Nuclear Service Center (WNYNSC) near West Valley, New York, and for incorporating the radionuclides in that waste into a solid.(a) This study was conducted by the Pacific Northwest Laboratory, with facility and equipment design by the Vitro Engineering Corporation, at the request of the United States Department of Energy as the West Valley Support Program. The high-level liquid waste accumulated from the operation of a chemical reprocessing plant by the Nuclear Fuel Services, Inc. from 1966 to 1972. The high-level liquid waste consists of approximately 560,000 gallons of alkaline waste from Purex process operations and 12,000 gallons of acidic (nitric acid) waste from one campaign of processing thoria fuels by a modified Thorex process (during this campaign thorium was left in the waste). The alkaline waste contains approximately 30 million curies and the acidic waste contains approximately 2.5 million curies.

The reference process described in this report is concerned only with chemically processing the high-level liquid waste to remove radionuclides from the alkaline supernate and converting the radionuclide-containing nonsalt components in the waste into a borosilicate glass. Cell preparation, which includes removal of existing reprocessing equipment and decontamination of the cell to allow installation of waste processing equipment with direct contact by construction personnel, is outside the scope of this report and is not included in this engineering and cost analysis. Other related activities not included in this report are the retrieval operations at the tank farm; preparation for and shipment of high-level waste packages; treatment, packaging and transportation of secondary wastes, mostly low-level wastes or transuranic

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(a) Glass is used as the reference waste form for the solidification process, because of the advanced stage of development of borosilicate glass monoliths. However, this choice does not imply a decision to actually use this waste form. An updated technical review of the waste form options will be prepared at the time the waste form is selected.

wastes arising from the processing of high-level radioactive waste; interim storage; and final decontamination and decommissioning of all equipment and facilities used in the solidification project. All of these related activities are the subject of study by other Department of Energy contractors. Those results, as well as the studies described in this report, will be published in a separate report.

1.1 FLOWSHEET SELECTION

The two most viable options for processing the high-level liquid waste at West Valley and converting it into a vitrified product are the salt/sludge separation process and the combined waste process. These two processes represent differing degrees of complexity.

In the salt/sludge separation process the insoluble fraction of the alkaline waste is separated from the soluble supernate fraction, a solution of salts--principally sodium nitrate/nitrite. The supernate is processed by ion exchange to remove most of the soluble radionuclides, i.e., cesium, strontium and actinides, which leaves the salt solution as a low-level waste. The heated salt solution is then concentrated by evaporation to about 20% water. The resulting salt cake is packaged for temporary storage onsite.

The radionuclides on the ion-exchange resin are removed and further processed to reduce the sodium salts carried through the ion-exchange process. The recovered radionuclides are mixed with the sludge and glass formers and are processed through a spray calciner/in-can melter system to fix the radioactive components of the waste in a borosilicate glass. The acidic Thorex waste is processed directly with glass formers through the spray calciner/in-can melter to a borosilicate glass. The glass is formed in a steel canister and is allowed to cool. The canister is then remotely welded shut for interim storage onsite.

In the combined waste processing approach all components of the alkaline waste (both soluble and insoluble portions) are combined with the acidic Thorex waste. The solid derived from this mixture with its high sodium nitrate/nitrite content has a relatively low melting point and would require several calcining systems for processing at required throughput by the spray calciner/in-can melter system. Therefore, a liquid-fed ceramic melter was chosen for this process. The mixed feed and glass formers are fed continuously to the liquid-fed ceramic melter. The mixture is maintained in a liquid state (molten glass) at high temperature by internal electrical resistance (commonly referred to as joule heating). The molten borosilicate overflows from the liquid-fed ceramic melter into canisters. The canisters are cooled and remotely welded shut for interim storage onsite.

The salt/sludge separation process was chosen for the reference process primarily due to its more advanced status of development and its lower overall costs. The reasons for this choice are:

- The salt/sludge separation process represents overall the more advanced state of development of the options considered, although some additional development and verification testing with WNYNSC waste is needed.
- The number of high-level waste canisters formed in the salt/sludge separation processing approach is approximately 16% of those formed by the combined waste approach. Transportation of the high-level waste canisters to a federal repository and disposal of the canisters in the repository represents a major cost factor.
- One vitrification system (spray calciner/in-can melter) would handle the waste through the salt/sludge separation process, while three vitrification systems (liquid-fed ceramic melter) would be required for the combined waste process, based on current technology.
- There are uncertainties associated with off-gas treatment and construction materials for the combined waste process.

Less favorable aspects of the salt/sludge separation process, compared to the combined waste process, include:

- a larger number (seven) of batch processes compared to two for the combined process
- a larger number of mechanical equipment pieces, although all have been used or demonstrated in radioactive service

- the potential difficulty in installing the many unit operations in existing shielded facilities of the reprocessing plant at WNYNSC.

On balance, it was judged that the salt/sludge separation process was the preferred choice for establishing a reference design.

1.2 DESCRIPTION OF THE REFERENCE PROCESS

The process is divided into eight functional systems. These systems are designated A through H as follows:

System	Function
A	Centrifugation
B	Agglomeration, settling, and filtration
C	Cesium ion exchange and concentration
D	Strontium ion exchange
E	Calcination, vitrification and canister handling
F	Recycle and secondary evaporation
G	Calciner effluent treatment (off gas)
H	Salt solidification

Most of the high-level waste processing equipment (System A through G, see Section 6.0) is located in the Chemical Processing Cell. The final salt solution concentration and packaging equipment (System H) is located in the Scrap Removal Room. Portions of the final process effluent cleanup equipment are located in the Process Ventilation Cell, which was provided by Nuclear Fuel Services but which has not been used.

A simplified flowsheet of the salt/sludge separation process, including the processing of the acidic Thorex wastes as well as the processing of the alkaline Purex waste, is shown in Figure 1.1.

The acidic Thorex waste will be the first waste processed, making its stainless steel underground tank available for use as a surge tank in handling the alkaline Purex waste. The acid stream is fed directly to the spray calciner to calcine the dissolved contents to oxides. The calcine is mixed with frit in the cone section of the calcination chamber and dropped into a

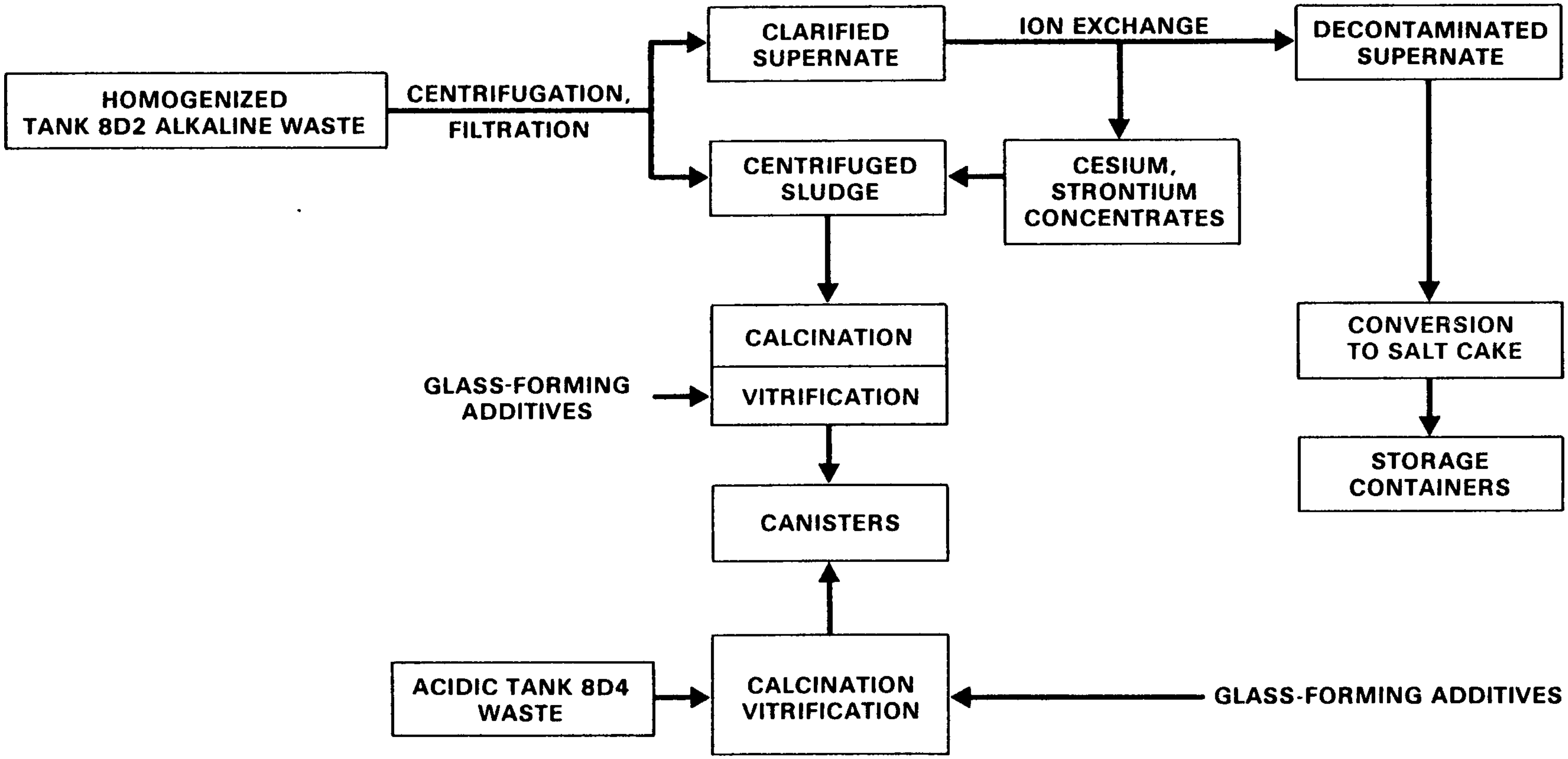

FIGURE 1.1. Simplified Flowsheet for the Salt/Sludge Separation Process Approach Including the Processing of Acidic Thorex Waste

stainless steel canister. The canister, which is in a furnace, is maintained above the melting temperature of the glass as it is being filled. After filling, the canister is removed from the furnace and cooled. A lid is then welded on remotely and the weld is helium leak-tested. The canister is then given gross decontamination and removed from the Chemical Processing Cell to the Equipment Decontamination Room for further handling.

The waste loading in the borosilicate glass produced from the salt/sludge separation process is about 25 wt%, but the glass produced from the acidic Thorex waste has a waste loading of less than 15 wt% due to the low solubility (about 10 wt%) of thoria in the glass.

In the reference salt/sludge separation method for processing the alkaline Purex waste, a slurry of alkaline sludge and supernate is received in the Chemical Processing Cell from the tank farm. The slurry is fed to a basket centrifuge to remove the bulk of the solids. Additional clarification is achieved by adding coagulation agents, allowing the solid to settle and filtering the solids using sand filters. The high separation of solids is necessary to prevent plugging of ion-exchange columns downstream and/or the carryover of radioactive fines which may pass through the ion-exchange systems.

The clarified supernate is fed to a Duolite® ion-exchange column to remove cesium. The effluent from this column is passed through other ion-exchange columns to sorb any soluble strontium and actinides. The processed supernate containing mostly sodium nitrate and nitrite salts, and only low-level concentrations of radionuclides, is then pumped to the Scrap Removal Room where the solution is concentrated by evaporation to form a salt cake containing approximately 20 wt% water. The salt cake is packaged in 55-gal drums and is moved to interim storage.

The cesium on the Duolite ion-exchange column is eluted with an ammonium hydroxide-carbonate solution. The sodium content of the eluate remains high. Therefore, after destruction of the elutriant by steam stripping and concentration in an evaporator, the concentrate is fed to a zeolite column where cesium is selectively absorbed. The sodium in the effluent is recycled. The zeolite

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containing the adsorbed cesium is added to the feed tank for the spray calciner. Strontium is eluted from the strontium ion-exchange column with dilute nitric acid and is also added to the spray calciner feed tank. Actinide ion-exchange resin may be either added to the spray calciner feed or packaged as transuranic waste without elution or regeneration.

The centrifuged sludge cake, settled solids, and backwashes of the sand filters, including the sand itself as necessary, are also added to the feed tank for the spray calciner.

The slurry of solids in the spray calciner's feed tank is fed through a spray nozzle at the top of the spray calciner. The droplets are calcined to an oxide as they drop through the calcination chamber. The oxide particles are mixed with glass formers (frit) in the cone section at the bottom of the calciner. The mixture drops into a stainless steel canister surrounded by an electrical-resistant furnace. The solids melt and form a borosilicate glass in the canister. These canisters of borosilicate glass made from the alkaline waste are subsequently treated in the same sequence of cooling and closure as those formed in processing the acidic Thorex waste.

Off gases from the calciner are filtered through stainless steel filters attached to the calciner and are further treated. The off-gas treatment system is designed to reduce the radionuclide and chemical content of the gas for discharge to the environment in compliance with the Environmental Protection Agency and the Nuclear Regulatory Commission standards.

1.3 DESIGN BASES

In addition to pertinent design and processing bases discussed above, the following items are inputs to the bases, criteria and/or assumptions for the design.

- The waste will be processed in a three-year period using a three-shift per day, seven-day week schedule.
- Existing cells in the reprocessing building will have been stripped of processing equipment and decontaminated for use with the high-level waste processing and solidification equipment. Cell modifications such as new penetrations will be undertaken as necessary to

allow high-level waste processing equipment to be installed. New facilities will not be required for the high-level waste processing.

- Existing basic support services are to be used to the extent practicable.
- Processing equipment shall be designed for remote operation and maintenance.
- Proven designs and state-of-the-art technology are used whenever possible.
- Canisters and their contents are expected to maintain integrity during processing, handling, interim storage and transportation to the repository.
- Criticality control is not required since there does not appear to be any credible mechanism by which criticality could occur during the processing of the waste with its low fissile concentration.
- Applicable codes and standards shall be applied.

1.4 CONCLUSIONS

It is feasible to install and remotely maintain and operate the reference salt/sludge separation processing and conversion equipment in existing cells of the reprocessing plant at the WNYNSC using the spray calciner/in-can melter system. These judgments are based on the engineering studies conducted by the Vitro Engineering Corporation and the resulting drawings presented in Appendix A. An arrangement of the equipment to be located in the Chemical Processing Cell is shown on drawings SK-7-961 through SK-7-967 and SK-7-980 and SK-7-981. An arrangement of the equipment to be located in the Scrap Removal Room is shown on drawings SK-7-969 and SK-7-970. The arrangement of the operating aisles is shown on drawings SK-7-958 through SK-7-960.

The construction and equipment cost estimate developed by the Vitro Engineering Corporation for the reference processing and vitrification facilities approaches $30 million without escalation. A breakdown of these costs is provided in Section 9.0. Detailed backup information is given in Appendix C. These costs are for the chemical processing of the high-level liquid waste and its conversion to a vitrified solid only. They do not include cell preparation, waste retrieval from the tanks, waste package preparation and transportation, repository operations, low-level radioactive wastes or transuranic waste handling and disposal, or decontamination or decommissioning.

Estimated manpower requirements for operating the reference processing and conversion facilities are 230 during startup and 145 after plant shake-down. These totals include operating personnel, supervisors, process engineers, and laboratory and other support personnel. Management, finance, purchasing personnel and other related support functions are not included in this total.

The selection of the salt/sludge separation process using the spray calcination/in-can melting system was based on previous plant and pilot-plant experience. This process is judged applicable to the high-level liquid waste stored at WNYNSC. Nevertheless, it is recommended that a supporting development program be identified early and conducted prior to plant implementation. These concerns apply particularly to experimental verification of the ion-exchange processes applied to the West Valley wastes and to potential improvements in processes that will reduce the process and equipment requirements. These studies may govern the overall timing of the project. The costs for these studies have not been estimated.

2.0 INTRODUCTION

The United States Department of Energy established the West Valley Support Program to provide process recommendations and a preconceptual design for the solidification of radioactive high-level liquid waste stored at the WNYNSC. The Department of Energy assigned management of the program to the Pacific Northwest Laboratory. This report presents a preconceptual design study for processing the high-level liquid waste and converting it to a vitrified product (borosilicate glass). The Vitro Engineering Corporation contributed the design details.

2.1 BACKGROUND

The high-level waste that is stored in underground tanks at WNYNSC was accumulated from 1966 through 1977 when Nuclear Fuel Services operated a fuel reprocessing plant there. About 625 tonnes of uranium fuel and 16 tonnes of thorium fuel were processed during this period (New York State Energy Research and Development Authority 1977). Approximately 30 million curies of fission products are in about 560,000 gallons of alkaline aqueous waste from the processing of uranium fuel, and about 2.5 million curies of fission products are in 12,000 gallons of acidic (nitric) aqueous waste from the processing of thorium fuel. The latter waste also contains the thorium, since there was no attempt to recover it.

In January, 1976, the United States Energy Research and Development Administration sent to the Nuclear Regulatory Commission a study prepared by the Pacific Northwest Laboratory, which examined alternative processes for managing high-level wastes at the WNYNSC (Nuclear Regulatory Commission 1976). Subsequently, the Argonne National Laboratory issued a more detailed study (Department of Energy 1978). These studies provided many of the bases used in this preconceptual design study for solidifying the existing wastes at WNYNSC.

2.2 BASIS AND SCOPE OF THE REPORT

The 1980 Energy and Water Development Appropriations Act directed the United States Department of Energy to provide necessary technical support to study and recommend a nuclear waste solidification program at West Valley, New York. Pacific Northwest Laboratory was given lead responsibility for performing a preconceptual design for a reference solidification process. The reference process that was chosen (as detailed in the main sections of this report) is the salt/sludge separation process for chemically treating the neutralized waste followed by solidification of the fraction containing the radionuclides using the spray calciner/in-can melter system. This process produces a vitrified product (glass) in a stainless steel canister. The acidic waste is processed directly in the spray calciner/in-can melter system. The preconceptual design discussed in this report: 1) considers the receipt of liquid (both acidic and alkaline) waste at the existing processing building from the underground storage tanks, 2) describes the treatment of the waste to reject most of the nonradioactive constituents, 3) defines the conversion of the radionuclide-containing concentrate to glass, 4) identifies the low-level wastes produced during the processing, 5) characterizes the off gases produced and their treatment, 6) describes the processing equipment and facilities and 7) presents the facility usage and layout.

2.3 TECHNOLOGY STATUS

The preconceptual design for the vitrification process proposed for use with the West Valley high-level waste is based on technology being developed and demonstrated in laboratory and pilot-plant facilities at Hanford (vitrification; off-gas processing; ion-exchange) or at the Savannah River Laboratory (sludge/supernate separation; ion-exchange processes).

2.4 RELATED DESIGN REPORT

A broader preconceptual design report will summarize the information presented in this report and will describe the additional activities necessary to complete the solidification project and to decontaminate and decommission the facilities used, items that are only peripherally treated in this report. The complete scope of the West Valley Demonstration Project to be covered in the broad report includes:

- preparation of the process cell by removing the existing equipment and decontaminating the cell;
- retrieval operations at the tank farm;
- transfer of the waste from the tank farm to the processing building;
- preparation of the packaged high-level waste for shipment, including placement of the canister in an overpack when necessary for transportation offsite;
- treatment and packaging of secondary wastes, mostly low-level wastes or transuranic wastes, arising from the vitrification process and supporting operations;
- movement of waste packages, both high-level waste and low-level wastes, to interim-storage facilities;
- miscellaneous support for facilities, utilities and construction;
- transportation of the waste from the site to a federal repository (high-level waste and transuranic waste) or to an approved burial ground (low-level wastes);
- safeguards and security operations;
- final decontamination and decommissioning of all equipment and facilities used in the demonstration project.

3.0 REFERENCE PROCESS SELECTION

This section summarizes the process and equipment considerations that were evaluated when selecting and defining the reference process for the preconceptual design study. A borosilicate glass was chosen as the reference waste form because of its advanced state-of-the-art technology. Implementation of the process using existing WNYNSC facilities is a goal.

3.1 PROCESS AND EQUIPMENT CONSIDERATIONS

Early in 1980 the United States' Department of Energy organized a panel of waste management experts to evaluate potential waste forms and processing options for immobilizing the high-level wastes at WNYNSC. This panel reviewed earlier studies (Nuclear Regulatory Commission 1976; Department of Energy 1978). Two processing approaches were considered to have sufficient technical merit to warrant further study. In the first approach, called the salt/sludge separation process in this report, the sludge and supernate of the alkaline Purex waste are separated. Radionuclides are then removed from the supernate and are solidified with the sludge into a vitrified waste form. The salt in the supernate is solidified into a salt cake by evaporation. In this first approach, the acidic Thorex waste is processed separately. In the second approach, called the combined waste process in this report, the acidic Thorex waste and the alkaline Purex waste are combined, including the soluble salts, and are processed together into a vitrified waste form.

In parallel with the assessment of the two processing options, equipment for performing the vitrification steps was assessed to determine the applicability of available vitrification systems to these processing options. The spray calciner/in-can melter was selected for the salt/sludge separation process and the liquid-fed ceramic melters for the combined waste process. These choices accommodate the unique characteristics of the high-level radioactive liquid waste resulting from each waste processing approach.

The two processing options were examined in detail as part of the West Valley Demonstration Project (Holton 1981). The requirements for the processes are outlined in the next subsection. This is followed by a brief description of each process.

3.1.1 Process Requirements

The process selected for the solidification of the high-level wastes stored at WNYNSC is to be based on the following process considerations:

1. All the alkaline Purex waste stored in Tank 8D2 and acidic Thorex waste stored in Tank 8D4 will be solidified.(a)
2. The high-level nuclear wastes will be immobilized in a three-year campaign.
3. The on-line efficiency for the waste immobilization facility operations will be 0.60. This efficiency factor includes processing all transfer line flushes, equipment washes and other liquid streams required to solidify the high-level waste.
4. The immobilization process will be capable of processing solutions used to decontaminate high-level waste storage tanks (principally oxalic acid), although this processing will occur after the three-year, high-level waste processing period.
5. Gaseous and liquid effluents produced from processing operations will meet release limits set by the Environmental Protection Agency and National Air and Water Quality Standards.
6. The high-level waste storage canister will be 2 ft in dia by 10 ft long. It will be filled with the waste product to approximately 8.5 ft, which represents a 614-L volume.
7. Existing technology will be used to the extent possible to define the process flowsheet and to develop the material balances.

(a) The chemical composition and properties of the alkaline Purex waste are summarized in Tables 5.1, 5.2 and 5.4. The chemical composition and properties of the acidic Thorex waste are summarized in Tables 5.3 and 5.4.

3.1.2 The Salt/Sludge Separation Process

A simplified schematic of the salt/sludge separation process is shown in Figure 3.1. The Thorex waste is homogenized and directly transferred to the high-level waste solidification equipment. This Thorex waste is solidified without any pretreatment. The alkaline Purex waste is solidified in a second campaign. Homogenized alkaline sludge and supernate are separated by centrifugation, gravity settling and filtration. The clarified supernate is treated by ion exchange to remove the radiochemical activity. The processed supernate is solidified by concentration into a low-level salt cake. The radionuclides recovered from the supernate are combined with the separated alkaline sludge and this waste blend is then fed to a solidification process.

Since the Thorex waste and the alkaline waste blend (the two waste streams to be solidified with the salt/sludge separation process) both have high melting points, they may be solidified using essentially any vitrification process. The spray calciner/in-can melter process was selected because it is the most highly developed vitrification process in the United States. The spray calciner/in-can melter system, which has been demonstrated on a radioactive pilot-scale and a nonradioactive full-scale, is considered essentially ready for plant application (Larson 1980). The low solubility of thoria in borosilicate glass was another consideration in the selection of the spray calciner/in-can melter system. If thoria should separate from the borosilicate glass as it is being formed in the canister, there would be no serious processing complications. On the other hand, if thoria precipitation should occur in the liquid-fed ceramic melter, removal of the precipitated thoria from the liquid-fed ceramic melter would present difficulties.

3.1.3 The Combined Waste Process

A simplified schematic of the combined waste processing approach is shown in Figure 3.2. This approach involves little waste pretreatment; however, there is more high-level waste to process than with the salt/sludge separation process since all the inert materials in the tanks, including the sodium salts, are incorporated into the glass.

In the combined waste process, alkaline Purex and acidic Thorex waste are blended, resulting in the neutralization of the Thorex waste and, possibly, in

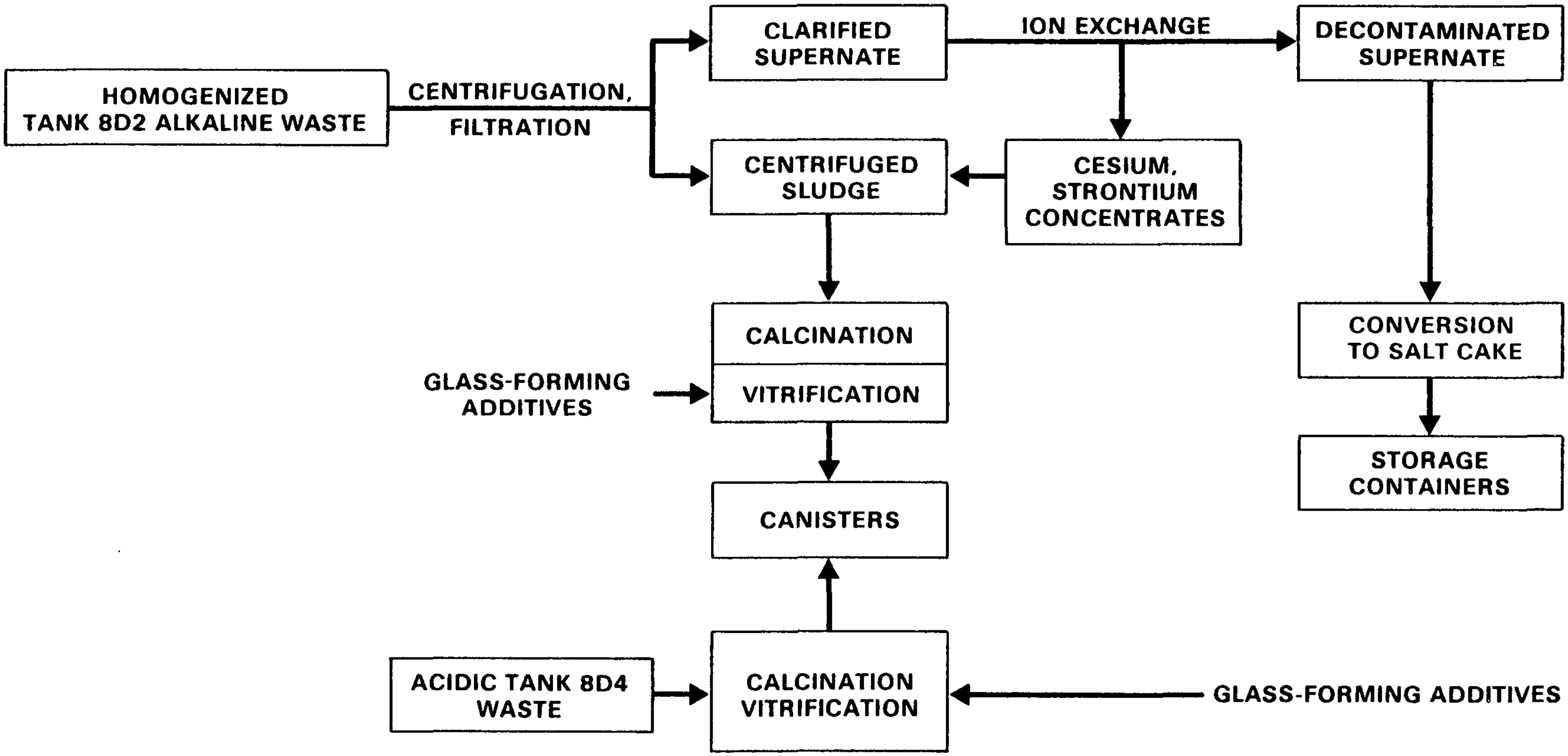

FIGURE 3.1. Simplified Flowsheet for the Salt/Sludge Separation Process Approach, Including the Processing of the Acidic Thorex Waste

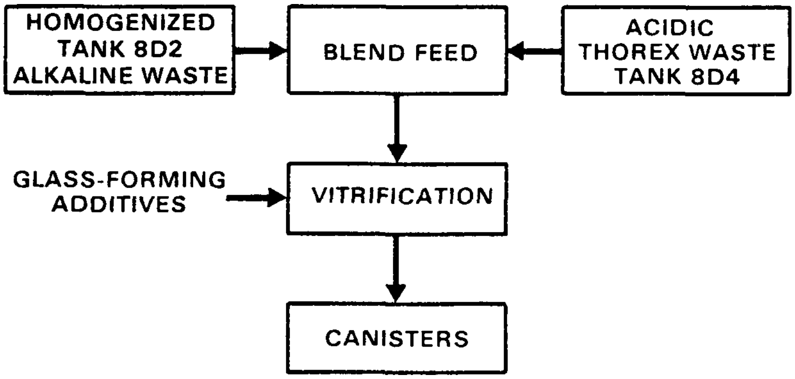

FIGURE 3.2. Simplified Flowsheet for the Combined Waste Process Approach

the precipitation of some metallic hydroxides. The waste mixture is then blended with glass-forming chemicals and is diluted with water. The neutralized waste blend is then transferred to a solidification process.

The combined waste processing approach incorporates both the radioactive and nonradioactive chemicals into the waste form. Most of the nonradioactive composition is sodium nitrate/nitrite. Sodium oxide is the limiting chemical constituent in the glass waste loading, which means only 18 to 20 wt% of equivalent waste oxides can be incorporated into the borosilicate-glass waste form. Thus, the amount of high-level waste product produced is six to seven times greater than that from the salt/sludge separation process in which most of the nonradioactive chemicals are removed before solidification.

The high concentration of sodium nitrate/nitrite in the combined waste blend reduces the melting point of the separated solids to between 300°C and 320°C. This low melting temperature would require lowering the operating temperatures of the spray calciner thereby significantly reducing the capacity of the spray calciner/in-can melter system. Therefore, multiple spray calciner/in-can melter systems would be required to meet required production rates. On the other hand, the liquid-fed ceramic melter system would be able to process this alkaline waste blend without undue capacity limitations (although it is estimated that up to three liquid-fed ceramic melters would be needed to process the combined waste in the projected three years). Therefore, the liquid-fed ceramic melter process was chosen as the reference vitrification process for the combined waste flowsheet. The liquid-fed ceramic melter process is under development and is expected to be ready for plant application within the next five years.

3.1.4 Process Equipment Alternatives

Successful development of the liquid-fed ceramic melter process will make the liquid-fed ceramic melter a viable alternative to the spray calciner/in-can melter in the salt/sludge separation process. The feasibility of installing a liquid-fed ceramic melter in the processing building, in lieu of the spray calciner/in-can melter system, is examined briefly in Section 6.0, Process Equipment. However, the liquid-fed ceramic melter is not considered a part of the reference process.

For the combined waste process, successful development of a larger liquid-fed ceramic melter might make it possible to use one melter in place of several melters. However, this possibility was not considered in this report because 1) the combined waste process was not chosen as the reference process and 2) installing a larger unit in existing processing cells for remote maintenance and operation and for final decontamination and decommissioning would require further study.

3.2 COMPARISON OF THE PROCESSES

The combined waste and salt/sludge separation processes (including their respective vitrification processes) were compared on the following factors (Holton 1981):

- process flowsheet and material balances
- status of technology of the complete process
- implementation of the process and equipment using existing WNYNSC facilities
- factors that dominate WNYNSC project cost.

Tables 3.1 and 3.2 summarize the results of this comparison. The significant differences between the salt/sludge separation process and the combined waste process are the rate of borosilicate glass production and the number of high-level waste canisters produced. The combined waste process is the simpler

TABLE 3.1. Comparative Summary of the Process Parameters for the Two Processes

Parameter	Salt/Sludge Separation Process Alkaline Waste	 Thorex Waste	Combined Waste Process Blended Waste
Borosilicate glass production rate, kg/h	24	82	190
Number of high-level waste canisters(a) 1500		180	60
Activity per canister, Ci	1.7×10^5	3.4×10^4	2.2×10^4
Decay heat per canister, W	506	110	65
Waste loading, wt%	25	15	19
Low-level waste salt cake, number of 55-gal drums	5120	--	--
Processing campaign length, d	597	60	657

(a) Glass weight per canister = 1993 kg. These quantities do not provide for contingencies such as partially filled canisters.

approach, but requires a large glass production rate, and therefore three liquid-fed ceramic melters of the current state-of-the-art design to accomplish the waste immobilization task within a three-year solidification campaign. The multiple liquid-fed ceramic melters, canister processing and handling equipment also increase the cost of the process. There are also major uncertainties associated with treatment of the off gases and ensuing waste (e.g., sulfates) from the combined process. Although the salt/sludge separation process appears to be more complex because of the processing required to separate the radionuclides from the alkaline supernate and to solidify the bulk of the nonradioactive neutral waste constituents as a salt cake (and the many batch processes involved), all of the process steps have been demonstrated. Simplification of the salt/sludge process is also thought to be possible.

The overall costs associated with the salt/sludge separation process as defined are lower than those associated with the combined waste processing approach (Holton 1981). This difference is the result of the high costs associated with packaging, interim storage, transportation and disposal of the

TABLE 3.2. Subjective Evaluation of the Two Processes

Items	Salt/Sludge Process	Combined Waste Process
Number of equipment pieces	70 (one vitrification system)	30 (three vitrification systems)
Types of batch processes	Centrifugation; gravity settling; sand filtration; cesium ion exchange; strontium ion exchange; waste blending; canister handling	Feed blending; canister handling
Number of mechanical equipment pieces	High, but routinely used in radioactive environment	Low
Process complexity	High, but demonstrated at other sites	High--three liquid-fed ceramic melters plus canister handling facilities and off-gas uncertainties
Status of development	High, but requiring verification testing	Low
Relative overall waste management cost	Lower	Higher--high-level waste packaging, transporation and disposal costs dominate.

high-level waste canisters in a federal repository. The salt/sludge separation process converts the bulk of the nonradioactive waste constituents to a low-level waste salt cake that may be disposed of at lower cost either at the WNYNSC site or at a licensed burial ground.

The salt/sludge separation process, using the spray calciner/in-can melter for the vitrification step, was selected as the reference solidification process for this preconceptual design study due primarily to the status of development and the lower overall waste management cost over the life cycle of the project when compared to the combined waste processing approach. Section 5.0 describes the salt/sludge separation process in more detail.

4.0 BASES FOR THE PRECONCEPTUAL DESIGN OF THE VITRIFICATION PROCESS

The overall West Valley Demonstration Project, as noted in the Foreword and Introduction, is concerned with demonstrating solidification techniques that can be used for preparing high-level waste for disposal. The project scope includes: 1) the removal of high-level liquid waste from underground tanks at WNYNSC, 2) the processing of the waste and the conversion of it to a solid suitable for transportation and disposal, 3) the disposing of low-level radioactive waste and transuranic waste produced by the solidification of the high-level waste, and 4) the decontamination and decommissioning of hardware and materials used in the project. The preconceptual design study described in this report is limited to that portion of the project concerned with the reference process for treating the liquid waste and converting it to a solidified product (borosilicate glass). This section presents the bases for the design of the processing and solidification equipment and its installation in the existing processing building at WNYNSC.

4.1 PROCESS CONSIDERATIONS

The process shall be based on the reference flowsheet described in Section 5.0. This flowsheet assumes that the acidic waste is processed directly into a borosilicate glass and that the supernate from the alkaline waste is processed through an ion-exchange system after which the concentrated radionuclides are combined with the sludge from the underground tanks for processing into glass.

Processing rates shall be based on receiving the waste solution from the tank farm in 4000- to 5000-L batches at a flow rate of approximately 380 L/min. The design capacity of all the equipment shall be 150% of nominal flowsheet rates. No provision shall be made for expanding the capacity of the process or equipment beyond these rates.

The preconceptual design of the vitrification process shall assure that effluents meet the Environmental Protection Agency's and Nuclear Regulatory Commission's environmental standards. Therefore, off gases shall be treated to reduce radioactive effluents released during normal operations to not more than 10% of limits established in applicable codes, standards and regulations.

The new iodine-removal system that was installed when Nuclear Fuel Services intended to modify the reprocessing facilities shall be modified as required to support the vitrification system.

The processing equipment shall be controlled automatically where practicable. Processing instrumentation required for safe operations shall be installed for the most part in the existing Control Room.

The sources, compositions and quantities of secondary wastes are to be identified. Treatment to a form suitable for shipment to an offsite federal repository (transuranic waste) or burial ground (low-level radioactive wastes) is not discussed in this report.

The processed salt solution (treated supernate) shall be pumped to the Scrap Removal Room where it will be dewatered by evaporation to a salt cake and packaged in 55-gal drums. The filled drums will be placed on a transfer device and moved out of the Scrap Removal Room. The subsequent handling and transfer of the drums to interim storage is a part of an interfacing design activity.

There does not appear to be any credible mechanism by which a criticality could occur in the equipment systems being considered for processing the high-level radioactive waste at WNYNSC. Therefore, no provision for criticality control has been considered for this process design. This preliminary conclusion will need to be examined after later safety analyses are done.

4.2 FACILITY CONSIDERATIONS

New equipment and facilities shall meet practices and standards for the Department of Energy's nuclear facilities. The solidification facilities for the high-level waste are to be subject to review andconsultation with the Nuclear Regulatory Commission. This review and consultation is to be conducted informally and will not include formal procedures or actions. For this design effort the Zone-2 earthquake code was used and the following design rationale was applied. The guidelines for modifying the facilities should be the same as those for new facilities, but as a minimum, the quality of the modification shall be as good as that of the existing facilities. Existing basic services shall be used to the extent practicable.

For the purposes of the preconceptual design and related cost estimates, it was assumed that the cell preparation shall have reduced the background radiation field to approximately 10 to 12 mR/h to permit contact installation of new equipment.

4.3 PRODUCT CONSIDERATIONS

The vitrified product shall be a borosilicate glass contained in a 304L stainless steel canister. Maximum dimensions of the sealed canister, including the lifting pintle, shall not exceed 24 in. in OD x 10 ft long.

The filled stainless steel canisters shall be welded closed, leak-tested with a helium mass spectrometer (as described in Section 5.3.6), cleaned of gross contamination, placed on a transfer device, and then moved from the Chemical Processing Cell to the Equipment Decontamination Room.

The canister and its contents shall maintain integrity throughout processing, in the course of normal handling, and during transportation from the site to the federal repository; it shall also meet all current legal requirements. This design effort does not include the capability of overpacking the canister.

It is assumed that process control of the spray calciner/in-can melter produces an acceptable waste form for a federal repository. It is also assumed that a limited number of canisters containing glass in which some calcine may have separated as a different solid phase will be acceptable at the federal repository. Therefore, verification of waste form homogeneity is not included. Final determination of this need awaits Environmental Protection Agency and Nuclear Regulatory Commission waste form criteria.

4.4 EQUIPMENT CONSIDERATIONS

The high-level waste solidification system shall be an integral part of the WNYNSC. All processing equipment, except final salt concentration and salt packaging equipment, will be installed in the Chemical Processing Cell.

Design features, similar to those required for fuel reprocessing plants, that facilitate decontamination and decommissioning shall be incorporated in the high-level waste vitrification facilities and processing equipment.

Existing piping shall be used to transfer the waste from the tank farm to the processing building. (Note that equipment layout drawings in Appendix A show new lines, an assumption made early in the design effort before it was determined that existing lines could be used.)

Proven designs and existing technology shall be used whenever possible. Innovations are to be minimized.

The equipment shall be designed where practicable for a minimum life of three years, the anticipated processing campaign period.

Processing equipment, piping, connections, in-cell instrumentation and associated materials shall withstand a radiation environment of 1×10^4 R/h during maintenance and continuous operation.

All processing equipment shall be designed for remote operation and maintenance. The need for and use of master-slave and/or electromechanical manipulators shall be minimized.

4.5 CODES AND STANDARDS

Applicable portions of the following codes and standards were also applied to the preconceptual design.

General

- Uniform Building Codes, latest edition
- National Fire Protection Association, latest edition
- National Electrical Code, latest edition
- Uniform Plumbing Code, latest edition
- ANSI N13.1-1969, Sampling Airborne Radioactive Materials in Nuclear Facilities

- ANSI N13.10-1974, <u>On-Site Instrumentation for Continuously Monitoring Radioactive Effluents</u>
- ANSI N2.3-1967, <u>Immediate Evacuation Signal for Use In Industrial Installations Where Radiation Exposure May Occur</u>
- ASME Section III, in accordance with Uniform Building Code earthquake, Zone 2.
- Occupational Safety and Health Standards
- National Electrical Manufacturer's Association
- American Refrigeration Institute Standards

<u>Department of Energy</u>

- Chapter 0505, "Construction Safety Program"
- Chapter 0510, "Prevention, Control and Abatement of Air and Waste Pollution"
- Chapter 0511, "Radioactive Waste Management and RL Appendix"
- Chapter 0512, "National Environmental Protection Act Implementation"
- Chapter 0513, "Effluent and Environmental Monitoring and Reporting"
- Chapter 0524, "Standards for Radiation Protection"
- Chapter 0544, "Planning for Emergencies in United States Energy Research and Development Administration Operations"
- Chapter 0550, "Operational Safety Standards"
- Chapter 0552, "Industrial Fire Protection"
- Chapter 6301, "General Design Criteria"
- Chapter 0820, "Quality Assurance Manual, RL Supplement and Appendix"

5.0 REFERENCE PROCESS DESCRIPTION

In this section the reference salt/sludge waste separation and solidification process is described in detail. The process bases, process flowsheets, material balances and operating philosophy are described for both the immobilization of the alkaline Purex waste and the acidic Thorex waste. Since the neutral waste will be processed and solidified separately from the acidic Thorex waste, the flowsheets for these two processes are presented separately.

5.1 GENERAL FLOWSHEET AND PROCESS DESCRIPTION

A simplified process flowsheet for the salt/sludge separation process is shown in Figure 5.1. This flowsheet is divided into eight separate unit processes. The purposes are to process and solidify the alkaline high-level waste and low-level waste side streams.

Homogenized alkaline waste is transferred from the waste tank farm to the centrifugation module. Here the bulk of the sludge is separated from the supernate by centrifugation. The centrate from the centrifuge is then transferred to a gravity settler and sand filters, and the sludge discharged from the centrifuge bowl is given three separate washes with water.

Each wash step includes a water wash of the centrifuge, agitation of the sludge/water mixture and separation of the homogenized slurry by centrifugation. The purpose of sludge washing is to remove soluble salts such as sodium nitrate and sulphate from the alkaline sludge to decrease the quantity of solids that must be vitrified. The washed sludge is transferred to the spray calciner feed-makeup tank, and the wash centrate is transferred to the recycle evaporation module.

Centrate from sludge/supernate separation is transferred to a gravity settler. The gravity settler further clarifies the supernate. Solids that collect in the gravity settler are recycled to the centrifuge feed tank. Supernate from the gravity settlers is then passed through two sand filters that remove essentially all of the remaining suspended solids. Clarification

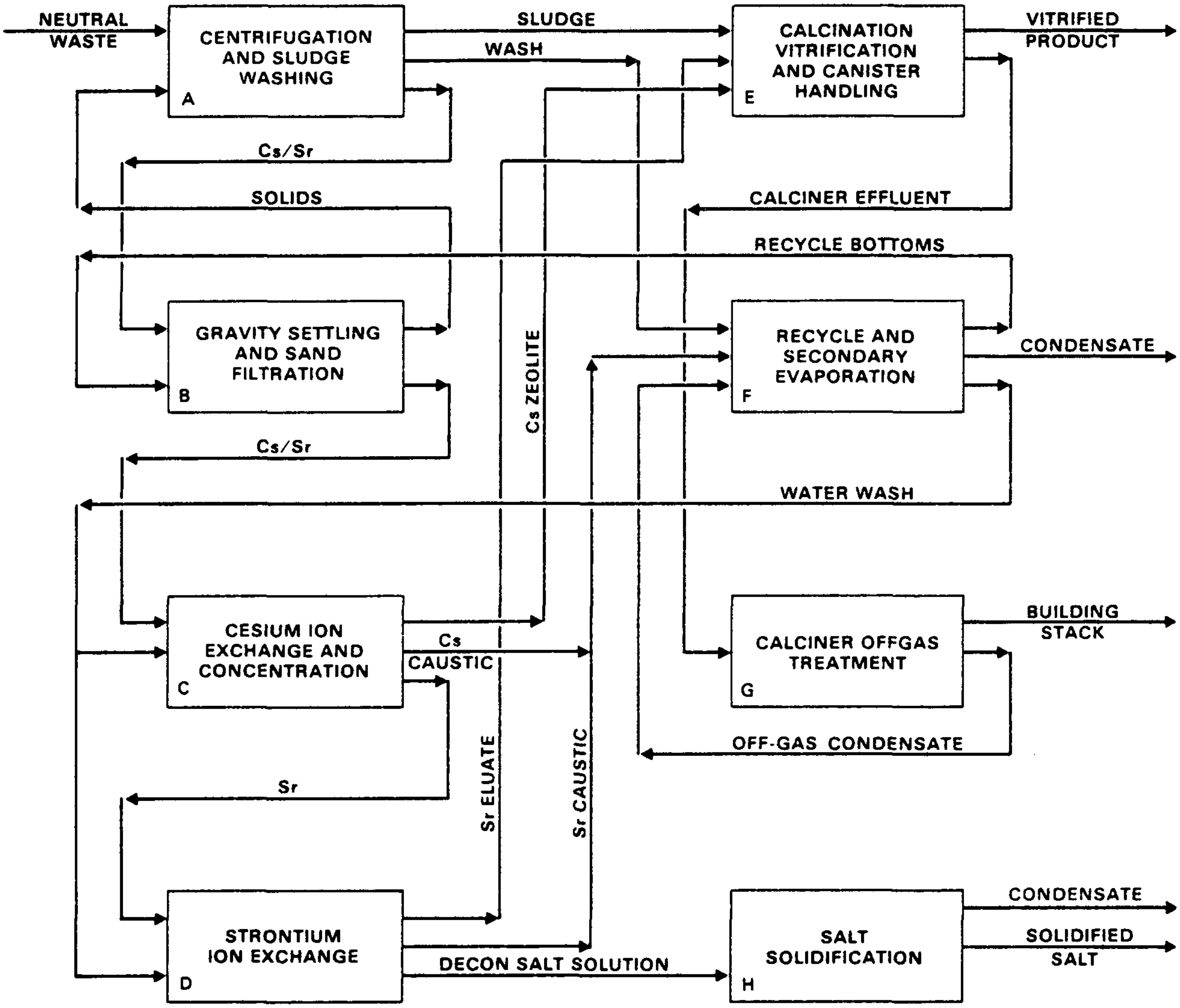

FIGURE 5.1. Simplified Process Flowsheet for Immobilization of Alkaline Sludge and Fission Products Recovered from the Alkaline Supernate

of the supernate to less than 1 ppm solids is necessary to prevent 1) radioactive solids from passing through the ion-exchange columns or 2) solids from plugging the ion-exchange columns.

The filtrate from the sand filters passes through ion-exchange columns. The first columns use Duolite ARC-359 ion-exchange resin to remove the cesium activity and most of the actinide activity from the alkaline supernate. Ion-exchange columns using a Chelex-100® ion-exchange resin removes the strontium and additional actinides. A backup ion-exchange column is included in the design in the event it should be necessary to remove any residual actinide activity. The resin for this backup exchange column has not been identified at this time.

Both the Duolite and Chelex ion-exchange columns are eluted and regenerated. The cesium eluate is concentrated and the ammonia and carbon dioxide elution agents are recovered. Since the sodium-to-cesium concentration ratio in the concentrator bottoms remains high, it is passed through a zeolite bed to selectively adsorb the cesium. The zeolite containing the adsorbed cesium is transferred to the spray calciner feed tank. Strontium eluate from the strontium ion-exchange columns is also transferred to the spray calciner feed tank. Solids from the backup ion-exchange column may be added to the spray calciner feed tank, or they may be packaged as transuranic waste.

The decontaminated salt solution from the ion-exchange columns is dewatered in a two-stage evaporation process, through a primary evaporator and through a wiped-film evaporator. Salt cake from the wiped-film evaporator is packaged in 55-gal steel drums. Condensates are processed through the low-level waste treatment facility.

The spray calciner feed consists of washed sludge, cesium-loaded zeolite, strontium eluate and miscellaneous solids such as spent sand filters. This feed is dried and simultaneously converted to oxides in the spray calciner. This calcine is then mixed with glass-forming chemicals (frit) and is melted to form a borosilicate glass using the in-can melting process. The canisters of high-level vitreous waste are 2 ft in dia by 10 ft high.

Effluents from the calcination/vitrification process include air, water, NO_x and volatile fission products. The radioactive off-gas treatment system includes scrubbers, condensers, filters and sorbers. Process scrub liquids are recycled to the evaporation module.

®Chelex is a registered tradename of Bio-Rad Laboratories.

The evaporation module consists of two evaporators in series. Bottoms from the first evaporator, the recycle evaporator, are recycled to the gravity settler; the overheads are sent to the second evaporator. Bottoms from the second evaporator are combined with miscellaneous waste streams and provide feed to the recycle evaporator; condensates are processed in the low-level waste treatment facility.

5.2 GENERAL PROCESS FLOWSHEET BASES

The general bases used in developing the flowsheet are presented in this subsection.

1. The instantaneous sludge slurry feed rate is 170 L/h. This rate will insure vitrification of the entire contents of Tank 8D2 (the alkaline waste storage tank) and Tank 8D4 (the acidic Thorex waste storage tank) in a three-year period assuming 60% onstream efficiency. The plant is assumed to operate on a four-shift, seven-day week schedule.
2. The flowsheet uses proven technology wherever possible. A conservative approach is assumed.
3. Spent ion-exchange resins from the cesium and strontium ion-exchange columns are considered low-level waste.
4. The supernate decontamination process reduces the concentration of cesium, strontium and actinides in the supernate to as low as reasonably achievable.
5. The decontaminated salt solution is evaporated to a solid containing 20% to 25% residual water.
6. The storage canisters are 2 ft in dia by 10 ft high and are filled with waste glass to 8.5 ft.
7. The waste loading in the borosilicate glass is 25 wt% for the alkaline waste and 15 wt% for the acidic Thorex waste.
8. The spray calciner/in-can melter process is the reference vitrification process.

9. The alkaline Purex waste in Tank 8D2 and the acidic Thorex waste in Tank 8D4 are treated separately.

10. The dilution of processing streams due to steam-jet transfers is not included in the material balances.

11. The alkaline waste is assumed to be completely homogenized before it is transferred to the Chemical Processing Cell. The volume of the alkaline waste is 2.12×10^6 L. Transfer line rinses add 1.89×10^5 L of water to the alkaline waste.

Table 5.1 lists the chemical composition of the alkaline Purex waste sludge and supernate. Table 5.2 gives the radionuclide inventory of the alkaline waste. The radionuclide data have been computed from the irradiation history of the fuels that were processed by Nuclear Fuel Services, Inc. The chemical inventory is based on Nuclear Fuel Services' records.

Table 5.3 summarizes the chemical and radionuclide composition of the acidic Thorex waste. The chemical inventory is based upon a laboratory chemical analysis performed by Oak Ridge National Laboratory. The results from this chemical analysis differ from chemical inventory records kept by Nuclear Fuel Services, Inc. This is particularly evident when reviewing the thorium inventory. This discrepancy is probably due to a nonrepresentative waste sample and/or precipitation of the thoria on the tank bottom. The radionuclide content was computed from the irradiation history of the thoria fuels.

Table 5.4 summarizes the estimated volumes, weights, radioactivity, and heat emissions for the supernate and sludge in tank 8D2 containing alkaline Purex waste and in tank 8D4 containing acidic Thorex waste. All radionuclide data has been extrapolated for decay to 1987, the assumed date to start waste processing. The data in Tables 5.2 through 5.4 were used to determine material and radionuclide balances for this study.

TABLE 5.1. Estimated Composition of Alkaline Waste in Tank 8D2(a)

Compound	Quantity, kg	
	Sludge	Supernate
$(Na,K)_2$ SO_4	---	89,800
(Na,K) NO_3/NO_2	---	1,090,000
(Na,K) OH	---	15,400
(Na,K) Cl	---	500
$Fe(OH)_3$	49,500	---
$FePO_4$	28,500	1,250
$Cr(OH)_3$(b)	3,900	2,100
$Ni(OH)_2$(b)	1,700	750
$Al(OH)_3$	830	---
AlF_3		1,570
Rare earths $M(OH)_3$(c)	1,620	---
Other fission products		
MSO_4	1,240	---
$M(OH)_4$	1,580	---
M_2O_3, MO_x^{-1}	210	220
Mo (as Na phosphomolybate, $Na_3PO_4 \cdot 12MoO_3$)	980	500
Mn (as MnO_2)[d]	2,200	---
Pu (as PuO_2)	35	---
U (as $Na_2U_2O_7$)	6,700	---
Actinides (total as oxides)	19	---
TOTAL	99,000	1,200,000

(a) Based on composite of information from Duckworth (1978) and calculations based on total MWd burnup in fuel processed at Nuclear Fuel Services, Inc.

(b) Assuming ratio of chromium to nickel to be 19:9 (the ratio in which it occurs in 304 stainless steel).

(c) Average atomic weight = 143.9 g/mol.

(d) Estimated.

TABLE 5.2. Radionuclide Inventory of Alkaline Waste in Tank 8D2(a)

Fission Products		Actinides	
Radionuclide	Quantity, Ci	Radionuclide	Quantity, Ci
Se-79	50	Np-237	23
Sr-90	6.7×10^6	Np-239	220
Y-90	6.7×10^6	Pu-238	1,500
Zr-93	250	Pu-239	1,800
Nb-93m	240	Pu-240	970
Tc-99	1900	Pu-241	70,000
Ru-106	110	Pu-242	1
Rh-106	110	Am-241	20,000
Pd-107	6	Am-242	180
Sb-125	6,100	Am-242m	180
Te-125m	6,100	Am-243	220
Sn-126	40	Cm-242	1
Sb-216m	40	Cm-244	8,800
Sb-126	40	Cm-245	1
Cs-134	21,000	Cm-246	1
Cs-135	35	TOTAL	1.0×10^5
Cs-137	8.9×10^6		
Ba-137m	8.4×10^6		
Ce-144	11		
Pr-144	11		
Pm-147	61,000		
Sm-151	200,000		
Eu-152	410		
Eu-154	130,000		
TOTAL	3.1×10^7		

(a) Extrapolated to 1987.

TABLE 5.3. Chemical and Radionuclide Inventory of Tank 8D4(a)

Constituent	Quantity
Element, kg(b)	
Fe	2,200
Cr	540
Ni	320
K	150
Al	460
Cl	63
S	32
Mn	32
B	18
Ca	6.8
Na	310
Th	10,300
Sm	9
Nd	45
Pm	1.4
Ce	16
Pr	8.1
La	8.1
Cs	32
Ba	23
Pd	4.5
Rh	5.4
Ru	16
Mo	25

(a) Based on information from Charles Luner (Argonne National Laboratory).

(b) Based on chemical analyses performed by Oak Ridge National Laboratory on 9/23/78 (Department of Energy 1978; Table 3.12).

(c) Extrapolated to 1987.

TABLE 5.3. (Contd)

Constituent	Quantity
Element, kg[(b)]	
Tc	8.1
Zr	5.4
Sr	8.1
Y	3.6
Rb	3.6
Se	2.7
As	0.5
Zn	3.6
Cu	2.7
Co	1.4
V	0.9
Ti	6.8
P	9.9
F	0.9
Radionuclide, Ci[(c)]	
Co-60	1,500
Sr-90	650,000
Y-90	650,000
Cs-134	540
Cs-137	690,000
Ba-137m	640,000
Eu-154	4,200
TOTAL	2.6×10^6

(a) Based on information from Charles Luner (Argonne National Laboratory).

(b) Based on chemical analyses performed by Oak Ridge National Laboratory on 9/23/78 (Department of Energy 1978; Table 3.12).

(c) Extrapolated to 1987.

TABLE 5.4. Summary of Estimated WNYNSC Wastes Extrapolated to 1987(a)

Waste	Volume, L	Weights, MT		Curies	Heat	
		Solids(a)	Water		Watts	BTU/h
Tank 8D2						
Supernate	2,100,000	1,200	1,500(c)	1.7×10^7	41,000	140,000
Sludge	28,000(d)	99	--	1.4×10^7	50,000	170,000
TOTAL	2,100,000	1,300	1,500	3.1×10^7	91,000	310,000
Tank 8D4	46,000	40	42	2.4×10^6	6,700	23,000
TOTAL	2,200,000	1,340	1,600	3.4×10^7	98,000	330,000

(a) Based on information from Charles Luner (Argonne National Laboratory).

(b) In the chemical forms assumed in the tables, complete calcination would reduce the total 700 MT, chiefly due to the $NaNO_3$-to-Na_2O conversion.

(c) Measured density of 1.3 taken for 8D2. Density of 1.8 assumed for 8D4.

(d) Assumed dry weight density = 3.5. About 6700 kg of this are thought to be uranate and 84 kg (equivalent to about 63,000 Ci and 1100 W) to be transuranic oxides.

5.3 DETAILED PROCESS FLOWSHEETS--SOLIDIFICATION OF THE ALKALINE PUREX WASTE

The salt/sludge separation process is described in detail in this section. The process flowsheet has been divided into eight separate unit operations as noted in Figure 5.1. These are:

A. centrifugation and sludge washing (shown in Figure 5.2)

B. gravity settling and sand filtration (shown in Figure 5.3)

C. cesium ion exchange, cesium eltriant recovery and cesium zeolite fixation (shown in Figures 5.4 and 5.5)

D. strontium ion exchange (shown in Figure 5.6)

E. calcination, vitrification and canister handling (shown in Figure 5.7)

F. recycle and secondary evaporation (shown in Figure 5.8)

G. calciner off-gas treatment (shown in Figure 5.9)

H. salt solidification (shown in Figure 5.10).

Each of the above unit operations is described separately below. Each description includes the purpose of the unit operation, a flow diagram of the unit operation, the bases and assumptions used in the material balance, the material balance, and the status of technology. The process flowsheet for each of the unit operations was developed from existing waste management technology. This technology was developed by the Department of Energy laboratories for high-level wastes at the Hanford and Savannah River sites and for commercial waste.

5.3.1 Centrifugation and Sludge Washing (System A)

The purpose of centrifugation is to separate more than 95 wt% of the solids from the homogenized sludge/supernate mixture. The centrifuge sludge is washed to remove essentially all of the soluble sodium salts. Washing reduces the amount of solids that must be vitrified.

Process Flowsheet

Figure 5.2 shows the flowsheet for the sludge/supernate centrifugation and sludge washing processes. Homogenized alkaline waste slurry containing both sludge and supernate is transferred to the feed holding tanks from the tank farm and is mixed with skim liquor from the primary centrifuge, bottoms from the gravity settler, and backflush from the sand filter. The resulting centrifuge feed mixture is sampled before being sent to the primary centrifuge. Samples are taken to estimate the centrifuge cycle time and feed rate.

The slurry is fed first to the primary centrifuge where approximately 90 wt% of the solids are removed. The centrate is passed through a second centrifuge where approximately 50% of the remaining solids are removed. Skim liquor from each centrifuge is recycled to each respective feed tank. The centrifuge cakes are mechanically discharged from each centrifuge and are combined with centrifuge wash water in the sludge slurry tank. The centrate containing soluble cesium and strontium from the secondary centrifuge is transferred to the gravity settler.

Sludge washing requires the use of the primary centrifuge. Additional wash water is added to the sludge slurry mixture in the sludge slurry tank to

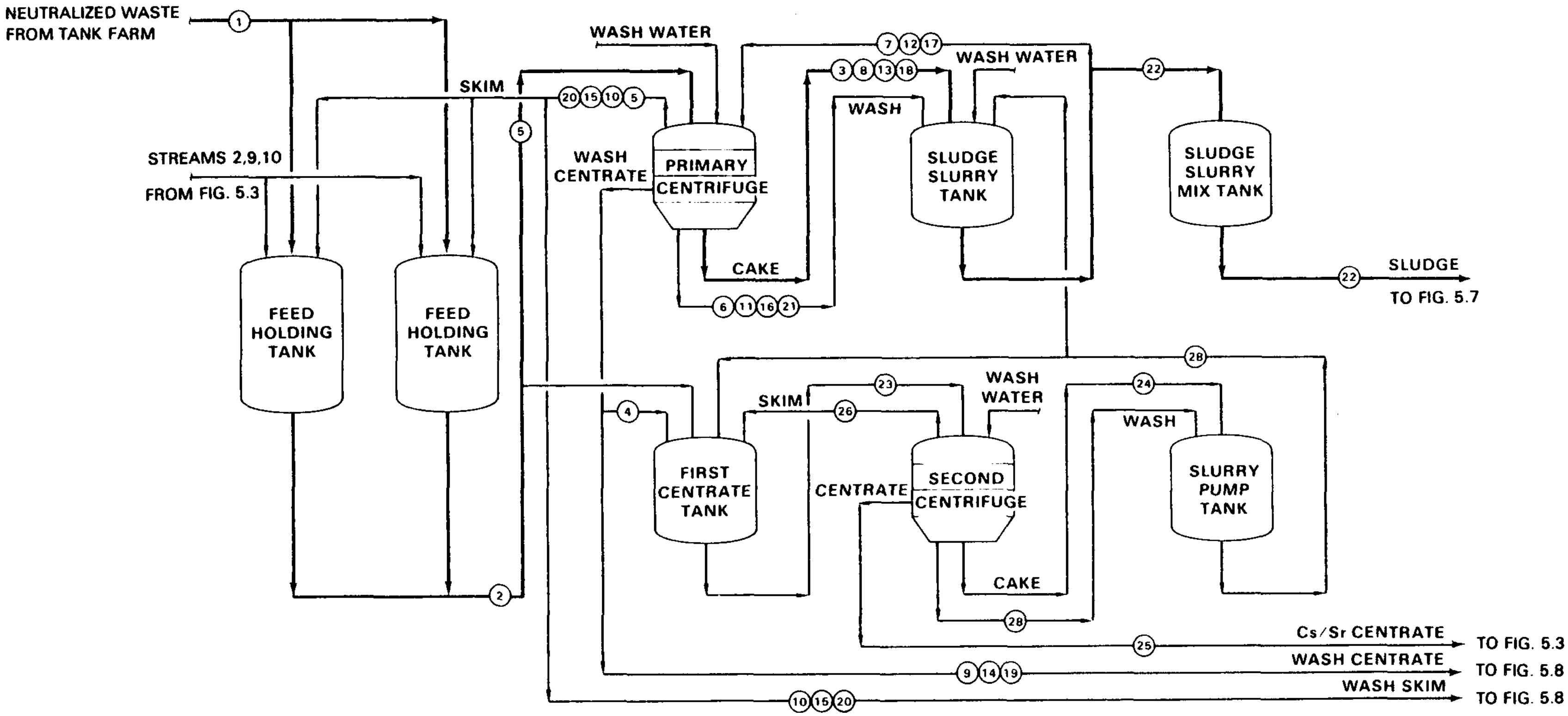

FIGURE 5.2. Flowsheet for Sludge/Supernate Centrifugation and Sludge Washing

adjust the soluble-salt-to-water ratio. This mixture is agitated and then fed to the centrifuge. Wash centrate and skim are transferred to the recycle evaporator. No solids are assumed to be in the cake skim. The cake is mechanically discharged into the sludge slurry tank. A water spray is used to further remove solids. The water is adjusted and the wash step is repeated three times. Approximately 85 wt% of the sodium, nitrate and sulphate is removed during each wash step.

Specifications given for the washed sludge for sodium, nitrate, and sulphate are ≤ 5 wt% Na^+, ≤ 10 wt% NO_3^- and ≤ 3 wt% SO_4^{-2} on a dry weight basis. Reducing the sodium concentration to the required level minimizes the quantity of vitreous product formed. Reducing the nitrate level in the sludge reduces the volatility of principle radionuclides such as ruthenium-106. The sulphate level in the sludge is reduced to prevent sodium sulphate phase-separation in the vitreous product. Sulphate solubility in borosilicate glass is less than 1 wt%. Concentrations greater than 1 wt% sulphate ion in the glass can lead to the formation of a water-soluble sodium sulphate in which other alkali elements such as cesium would tend to concentrate.

Material Balance

The bases for the material balance for sludge/supernate separation and sludge washing are presented below. The flowsheet conditions and material balance data are summarized in Table 5.5.

- The feed rate of homogeneous neutral waste to the centrifugation process is 170 L/h.
- Separation factors of 10 and 2 are assumed for insoluble solids across the primary and secondary centrifuges, respectively.
- The time cycle of the primary centrifuge is two hours; the time cycle of the secondary centrifuge is eight hours.
- The material balance for centrifuge operation assumes continuous rather than semibatch operation.
- The ratio of soluble salt to water is the same in the centrifuge cake as in the centrifuge feed.

TABLE 5.5. Flowsheet Conditions and Material Balance Data for Sludge/Supernate Separation and Sludge Washing

Operation	Primary Centrifuge Operation					
Stream Number	1	2	3	4	5	6
Variable / Description	Neutralized Waste From Tank 8D-2	Feed to Primary Centrifuge	Solids from Primary Centrifuge	Primary Centrifuge Centrate	Primary Centrifuge Skim	Primary Centrifuge Wash Water
Flow, kg/h						
Salt	83.84	96.96	9.41	84.73	2.82	--
Solid	6.91	8.147	7.118	0.791	0.237	--
H_2O	172.8	210.9	20.5	184.3	6.14	84.4
Total	263.5	316.01	37.02	269.8	9.20	84.4
Temperature, ^oC	35	35	35	35	35	35
Density, g/cm^3	1.55	1.50	1.80	1.47	1.5	1.0
Flow, L/h	170.0	211.0	21.0	184.0	6.1	84.4
Activity, Ci/L	12.69	11.98	54.2	7.15	12.0	--
Heat, W/L	0.037	0.035	0.19	0.018	0.036	--

TABLE 5.5. (Contd)

Operation	First Wash Cycle					
Stream Number	7	8	9	10	11	12
Description / Variable	First Cycle Solids Reycle	Solids Recycle	First Wash Cycle Centrate	First Wash Cycle Skim	First Cycle Wash Water	Second Cycle Solids Recycle
Flow, kg/h						
Salt	17.88	2.68	15.00	--	--	2.68
Solid	7.514	7.28	0.225	--	--	7.28
H_2O	182.64	10.03	167.1	5.54	85.0	95.0
Total	208.03	19.99	182.3	5.54	85.0	104.98
Temperature, ^oC	35	35	35	35	35	35
Density, g/cm^3	1.13	2.0	1.09	1.0	1.0	1.10
Flow, L/h	182.6	10.0	167.0	5.54	85.0	95.0
Activity, Ci/L	7.21	107.0	1.46	--	--	11.24
Heat, W/L	0.002	0.37	0.004	--	--	0.04

TABLE 5.5. (Contd)

Operation	Second Wash Cycle					Third Wash Cycle			
Stream Number	13	14	15	16	17	18	19	20	21
Description / Variable	Solids Recycle	Second Wash Cycle Centrate	Second Wash Cycle Skim	Second Cycle Wash Water	Third Cycle Solids Recycle	Solids Recycle	Third Wash Cycle Centrate	Third Cycle Wash Skim	Third Cycle Wash Water
Flow, kg/h									
Salt	.402	2.28	--	--	0.402	0.06	0.342	--	--
Solid	7.06	0.22	--	--	7.06	6.91	0.150	--	--
H_2O	7.54	84.26	3.2	86.8	94.36	6.91	84.62	2.83	39.1
Total	15.0	86.76	3.2	86.8	101.8	13.88	85.11	2.83	39.1
Temperature, ^oC	35	35	35	35	35	35	35	35	35
Density, g/cm^3	2.0	1.02	1.0	1.1	1.07	2.0	1.0	1.0	1.0
Flow, L/h	7.5	84.0	3.2	86.8	94.4	6.9	84.6	2.83	39.1
Activity, Ci/L	133.0	0.76	--	--	10.6	141.7	0.31		--
Heat, W/L	.48	0.002	--	--	0.038	0.51	0.001	--	--

TABLE 5.5. (Contd)

Operation	Secondary Centrifuge Operation						
Stream Number	22	23	24	25	26	27	28
Description / Variable	Sludge Slurry to Spray Calciner	Solids Feed to Secondary Centrifuge	Solids from Secondary Centrifuge	Secondary Centrifuge Centrate	Secondary Centrate Skim	Centrifuge Secondary Wash Water	Secondary Centrifuge Solids Recycle
Flow, kg/h							
Salt	.06	87.27	8.47	76.26	2.54	--	8.47
Solid	6.91	0.814	0.395	0.395	0.024	--	0.395
H_2O	39.1	189.8	1.84	182.47	5.53	75.9	77.74
Total	46.07	277.9	10.70	259.13	8.09	75.9	86.61
Temperature, ^oC	35	35	35	35	35	35	35
Density, g/cm^3	1.18	1.46	5.8	1.42	1.46	1.0	1.11
Flow, L/h	39.1	190.0	1.8	182.5	5.53	75.9	77.7
Activity, Ci/L	25.01	7.08	95.7	2.47	7.1	--	2.27
Heat, W/L	0.09	0.02	0.27	0.016	0.018	--	0.006

- Eighty-five wt% of the soluble salt in the sludge is removed during each wash.
- Water is added to the sludge slurry mixture to keep the solids in the feed at 6 to 8 wt%. The volume of the centrifuge feed is identical in each batch wash.
- Ninety-seven percent of the solids are removed in the centrifugation step.

State of Process Development

Centrifugal separation of sludge slurry mixtures similar to that proposed in this flowsheet has been performed in small, 1300-g, vertical, solid-bowl centrifuges with actual and simulated sludge slurries (Stone 1976). The results of these tests were similar for actual and simulated slurries. In both series of tests, cumulative sludge recovery in two centrifuges in series ranged from 90 to 99 wt%. The use of polyelectrolytes flocculants was shown to aid in separation in some cases; however, in general good separation could be achieved without the use of these agents.

Large-scale tests using a 48-in. batch basket centrifuge have also been performed (Landon 1980; Okeson 1980). In these large-scale tests 98 wt% of the solids were recovered. The test results also indicated that prescribed specifications for sodium, nitrate, and sulphate can be achieved.

Development work on the centrifuge for separation of alkaline sludge and supernate will continue at the Savannah River Laboratory in support of its Defense Waste Processing Facility (Jennings 1979).

5.3.2 Gravity Settling and Sand Filtration (System B)

The purpose of the gravity settling and sand filtration steps is to remove suspended solids from the centrifuge centrate. Removal of suspended solids prevents both the passage of radioactive solids through the ion-exchange columns (largely insoluble Sr-90) and the pluggage of the ion-exchange columns with larger suspended solids.

Process Flowsheet

The flowsheet for the gravity settling and sand filtration process is shown in Figure 5.3. Centrate from centrifugation is blended with recycle evaporator bottoms and sodium carbonate solution from the fixation of cesium on zeolite to form the feed for the gravity settler. Two gravity settlers are operated in parallel. An agglomeration agent is added and the resulting solution is heated to $90^{o}C$ and agitated, first vigorously and then gently. The agglomeration agent assists in the coagulation of the colloidal particles into heavier, faster-settling masses. After 10 to 15 minutes agitation is discontinued, the solution is allowed to cool and the agglomerated colloidal particles are allowed to settle. The clarified top layer, representing the majority of the feed volume, is decanted and transferred to the supernate holding tank. The bottom, settled sludge layer is left for resettling with additional feed batches. Several feed batches are received, settled and decanted before the settled sludge is transferred to the centrifuge sludge slurry feed tank.

A polyelectrolyte is added to the decanted supernate from the gravity settler which has been cooled to $25^{o}C$. The supernate is pumped through a sand anthracite filter and is collected in the filtrate holding tank. Solids removal in the first sand filter is 90%. Additional polyelectrolyte is added to the supernate which is then pumped through the polishing sand filter. Solids removal in the polishing sand filter is 80%.

Both the sand filter and the polishing sand filter are back-flushed daily to remove suspended solids. Backflush solutions are transferred to the centrifuge feed holding tank. The sand filters are fluidized monthly with air and caustic, and the fluidized slurry is transferred to the spray calciner feed-makeup tank.

Material Balance

The material balance bases not covered in the above description of the flowsheet for the gravity settling and sand filtration processes are summarized below. The material balance data for gravity settling and sand filtration are summarized in Table 5.6.

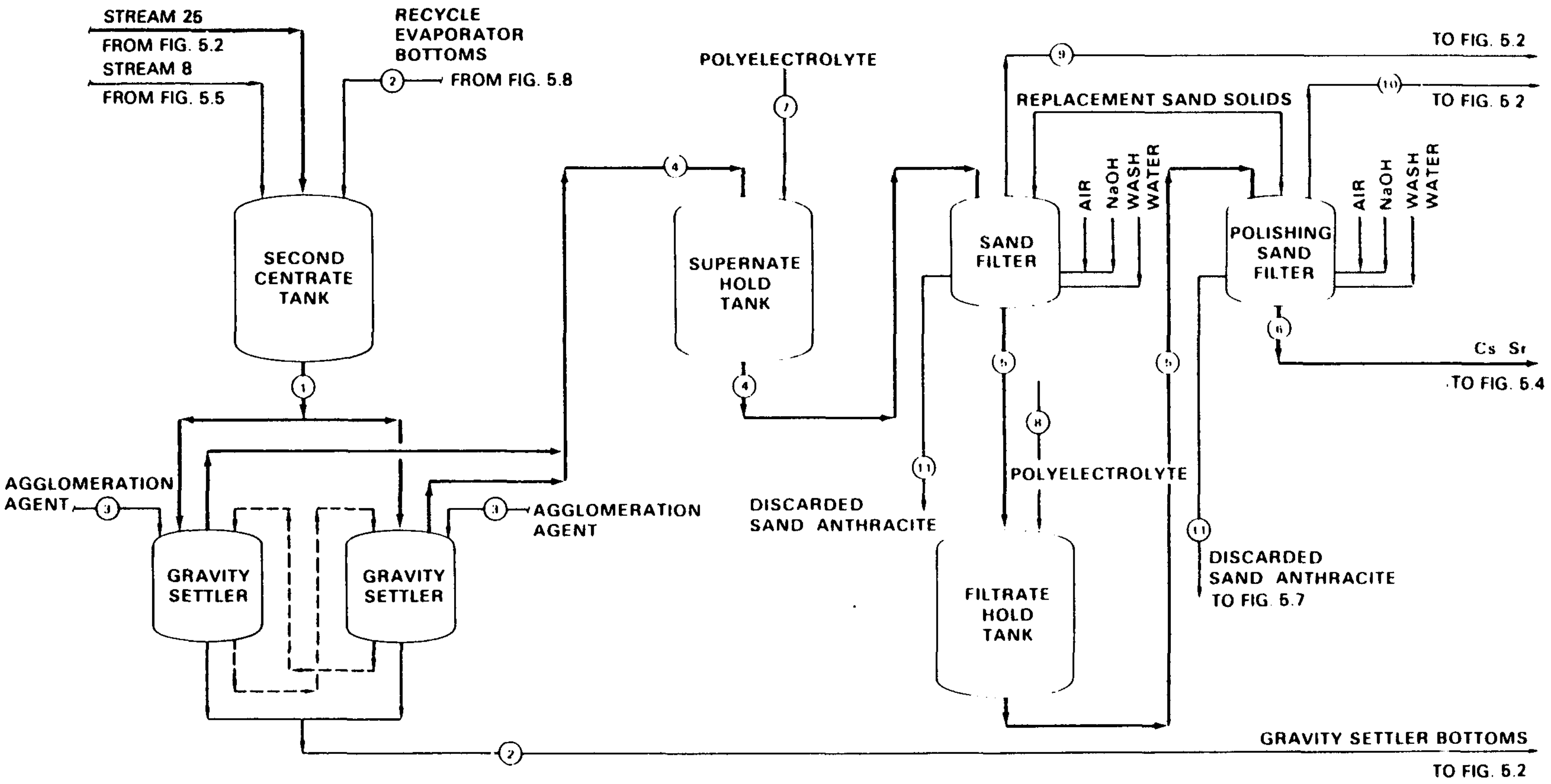

FIGURE 5.3. Flowsheet for Gravity Settling and Sand Filtration

TABLE 5.6. Flowsheet Conditions and Material Balance Data for Gravity Settling and Sand Filtration

Operation	Gravity Settling		
Stream Number	1	2	3
Description / Variable	Feed to Gravity Settlers	Solids from Gravity Settlers	Agglomeration Agent
Flow, kg/h			
Salt	101.76	10.3	0.003
Solid	1.00	0.98	
H_2O	262.54	17.0	0.03
Total	365.3	28.28	0.033
Temperature, ^oC	35	35	35
Density, g/cm^3	1.39	1.66	1.1
Flow, L/h	262.5	17.0	0.03
Activity, Ci/L	5.07	16.7	--
Heat, W/L	0.018	0.05	--

Operation	Sand Filtration				
Stream Number	4	5	6	7	8
Description / Variable	Supernate from Gravity Settlers	Filtrate from Primary Filter	Filtrate from Polishing Filter	Polyelectrolyte	Polyelectrolyte
Flow, kg/h					
Salt	91.46	91.46	91.46	0.38	0.38
Solid	0.02	0.002	0.0004		
H_2O	245.57	253.37	261.2	7.8	7.8
Total	337.05	344.83	352.66	8.18	8.18
Temperature, ^oC	25	25	25	25	25
Density, g/cm^3	1.37	1.36	1.35	1.05	1.05
Flow, L/h	245.6	253.4	261.2	7.8	7.8
Activity, Ci/L	4.86	4.69	4.56	--	--
Heat, W/L	0.011	0.011	0.011	--	--

TABLE 5.6. (Contd)

Operation	Sand Filter Backwash and Regeneration		
Stream Number	9	10	11
Description / Variable	Primary Filter Back Flush	Polishing Filter Back Flush	Discarded Filter Media(a)
Flow, kg/h			
Salt			
Solid	0.018	0.0016	0.64(b)
H_2O	7.5	7.5	
Total	7.5	7.5	0.64
Temperature, °C	25	25	35
Density, g/cm^3	1.0	1.0	2.0
Flow, L/h	7.5	7.5	--
Activity, Ci/L	0.34	0.03	--
Heat, W/L	0.001	0.0001	--

(a) Infrequent batch operation as required.
(b) 0.07 kg/h anthracite; 0.57 kg/h sand.

- Recovery of solids in the gravity settler is 98 wt%.
- One liter of 8 wt% starch solution is added to 8000 L of feed to aid settling.
- Both the primary sand filter and the polishing sand filter are backflushed with 180 L of process water daily to remove suspended solids.

Status of Technology

Large-scale gravity settling studies have been performed at the TNX Facility, Savannah River Laboratory (Landon 1980). In these studies, both high- and intermediate-caustic flowsheets were tested. The supernates tested were in the nominal compositional range expected for the WNYNSC flowsheet. These large-scale tests (6000-L gravity settler) have shown that less than 25 ppm sludge remains with the decanted supernate at 35°C to 45°C. Settling rates of 20 to 25 in./h were demonstrated. The tests have also shown that recycling the sludge several times in the gravity settler improves performance. The increased sludge loading produces a scavenging effect on the small particulates most difficult to settle.

Sand filter studies have been performed in a radioactive environment with actual clarified supernate at the Savannah River Laboratory (Stone 1976). In these tests centrate feeds containing 500 to 2000 ppm of suspended matter were clarified to less than 100 ppm of suspended matter.

Bench-scale and large-scale experiments have also been performed in a nonradioactive environment. The bench-scale tests with synthetic supernates produced a filtrate containing less than 1 ppm of suspended matter. The large-scale tests, conducted in a 17.25-in.-ID column, also produced a supernate containing less than 1.0 ppm of suspended matter when two filters were operated in series.

5.3.3 Cesium Ion-Exchange System (System C)

The first purpose of the cesium ion-exchange system is to remove essentially all of the cesium, plutonium and other actinides from the clarified supernate by ion exchange using a Duolite ARC-359 resin.

Process Flowsheet for Cesium Ion Exchange on Duolite

The flowsheet for the cesium ion-exchange process using Duolite ARC-359 is shown in Figure 5.4. Filtrate from the polishing sand filter is pumped through two ion-exchange columns in series containing Duolite ARC-359 ion-exchanger resin. Cesium, plutonium and other actinides are removed from the filtrate by the ion exchanger. The column effluent is transferred to the strontium ion-exchange feed tank. The loading step is followed by a water rinse downflow.

Cesium and the actinides on the ion exchanger are eluted with $2\underline{M}$ NH_4OH-$2\underline{M}$ $(NH_4)_2CO_3$ solution using upflow followed by a water rinse using upflow. Eluate and water rinse effluents pass on to elutriant recovery and cesium concentration processes. The ion-exchange columns are then regenerated with $2\underline{M}$ NaOH using upflow. Excess caustic is transferred to the recycle evaporator feed tank.

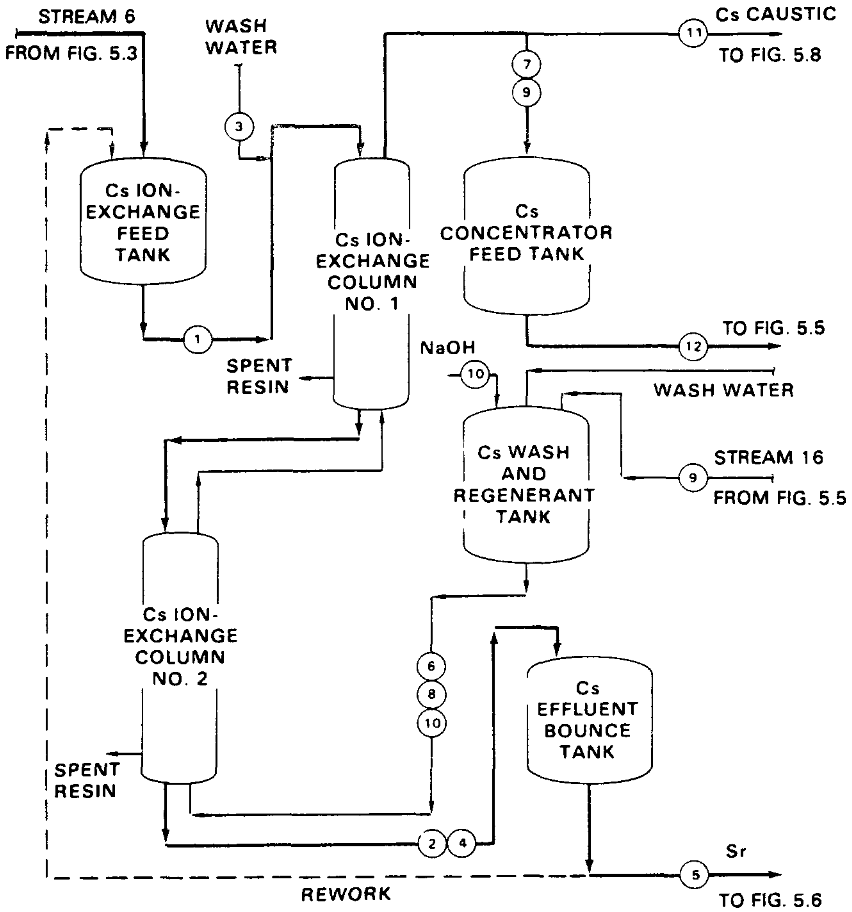

FIGURE 5.4. Flowsheet for Cesium Ion Exchange

Duolite ARC-359 resin is a strong-acid cation exchanger selective to cesium at high pH. Decontamination factors of 10^5 for cesium in actual clarified waste supernate and 300 for plutonium have been obtained (Wiley and Wallace 1975). Nominal decontamination factors of 10^4 for cesium and 100 for plutonium are used for material balance calculations.

A gamma detector placed between the two Duolite ion-exchange columns will continuously monitor Cs-137 in the first column effluent. The second column effluent can also be monitored; however, background radiation may cause too much noise. The effluent will be sampled and analyzed before it is transferred to the strontium ion-exchange process. Effluent containing more cesium than the specified concentration will be reworked by recycling it to the cesium ion-exchange feed tank.

Variations in the activity level and chemical characteristics of the column feed will exist. To compensate for these variations, the feed volume and cycle time can be adjusted to maintain the overall desired decontamination factors.

Material Balance for Cesium Ion Exchange

The bases used for developing the flowsheet and material balance for cesium ion exchange are summarized below. The material balance data are summarized in Table 5.7. The system consists of two ion-exchange columns in series packed with Duolite ARC-359 resin.

- The assumed resin stoichiometries are:

  loading--$RNa + Cs^+(aq.) = RCs + Na^+(aq.)$
  elution--$RCs + NH_4\ (aq.) = RNH_4 + Cs^+(aq.)$
  $RNa + NH_4\ (aq.) = RNH_4 + Na^+(aq.)$
  regeneration--$RNH_4 + Na^+(aq.) + OH^-(aq.) = RNa + NH_3(aq.) + H_2O$.

- The decontamination factors for cesium and plutonium are 10^4 and 350, respectively.

- Resin is replaced annually.

- Material balance rates are averaged over the cycle time of the resin bed.

- The time cycles and flow rates are:

Step	Time, h	Feed rate, Column Vol/h
Load	20	1.4
1st rinse	3	1
Elution	10	1
2nd rinse	3	1
Regeneration	5	1
Standby	3	1
Total	44	--

Status of Technology for Cesium Ion Exchange

Duolite ARC-359 resin has been used extensively at Hanford's B-Plant to remove approximately 95% of the cesium from Purex and REDOX waste streams.

TABLE 5.7. Flowsheet Conditions and Material Balance Data for Cesium Ion Exchange

Operation	Cesium Column Loading, 20 h/batch		Cesium Column Rinse, 3 h/batch		Effluent Collection
Stream Number	1	2	3	4	5
Description / Variable	Cesium Ion-Exchange Feed	Column Effluent	Column Rinse	Rinse Effluent	Feed to Strontium Ion Exchange
Flow, kg/h (kg)[(a)]					
Salt	(1644.8)	(1644.8)			(1644.8)
NaOH	(37.2)	(37.2)			(37.2)
Na_2CO_3	(107.2)	(107.2)			(107.2)
NH_4OH					
$(NH_4)_2CO_3$					
H_2O	(5224)	(5224)	(97.5)	(97.5)	(5321.5)
Solids	(.008)	(.008)			(.008)
Total	(7013.2)	(7013.2)	(97.5)	(97.5)	(7110.7)
Temperature, °C	25	25	25	25	25
Density, g/cm^3	1.34	1.34	1.0	1.0	1.30
Flow, L/h (L)	(5220)	(5220)	(97.5)	(97.5)	(5469.8)
Activity, Ci/L (Ci)	(23,650)	(94.0)	--	--	(87.5)
Heat, W/L	0.011	0	--	--	0

(a) Quantities in parentheses indicate batch flow; quantities not in parentheses indicate continuous flow rate.

TABLE 5.7. (Contd)

Operation	Cesium Column Elution, 10 h/batch		Cesium Column Rinse, 3 h/batch		Cesium Column Regeneration, 5 h/batch		Cesium Collection
Stream Number	6	7	8	9	10	11	12
Description / Variable	Cesium Eluant	Cesium Eluate	Column Rinse	Rinse Effluent	Caustic Regenerant	Cesium Caustic Eluate	Concentrator Feed
Flow, kg/h (kg)[a]							
Salt							
NaOH		(35)			(21.0)	(6.9)	
Na_2CO_3						(4.4)	4.66
NH_4OH	(72)	(57)		(2.28)			9.6
$(NH_4)_2CO_3$	(197)	(176)		(6.0)			15.4
H_2O	(850)	(850)	(97.5)	(97.5)	(256.0)	(256.5)	117.5
Solids							
Total	(1119)	(1118)	(97.5)	(105.78)	(277.0)	(267.8)	147.16
Temperature, °C	25	25	25	25	25	25	25
Density, g/cm^3	1.31	1.31	1.0	1.08	1.08	1.08	1.25
Flow, L/h (L)	(850)	(850)	(97.5)	(97.5)	(256.5)	(256.5)	117.5
Activity, Ci/L (Ci)		(139)	--	--	--	--	10.0
Heat, W/L	--	0.3	--	--	--	--	0.03

(a) Quantities in parentheses indicate batch flow; quantities not in parentheses indicate continuous flow rate.

Removal of Cs-137 from simulated and actual Savannah River Plant supernates has been demonstrated in laboratory tests using Duolite ARC-359 (Wiley 1976b; Wiley and Wallace 1975). The average decontamination factors in these tests were 4 x 10^5 for Cs-137 and 300 for plutonium. Decontamination factors for cesium of 10^4 were also demonstrated in supernates with sodium concentrations of 4.0<u>M</u> and 9<u>M</u>.

Process Flowsheet for Cesium Eluate Concentration

The process flowsheet for cesium elutriant recovery and cesium concentration is shown in Figure 5.5. Cesium ion-exchange elutriant is fed to a concentrator/steam stripper. Ammonia and carbon dioxide are stripped from the elutriant in the concentrator tower. Carbon dioxide and ammonia are condensed as the overhead products are collected in an eluate-makeup tank. Regeneration and additional preparation of the elutriant are accomplished by adding ammonium hydroxide solution and bubbling in carbon dioxide gas. A water scrub tower located over the eluate makeup tank removes volatile gases and returns them to the tank.

Water added during rinsing of the ion-exchange columns is removed through a side stream in the steam stripper. The steam removed in the sidestream is condensed in the purge condenser and is transferred to the recycle evaporator.

Material Balance for Cesium Eluate Concentration

The bases used to develop the material balance for elutriant recovery and cesium eluate concentration are summarized below. Table 5.8 gives the material balance data.

- Cesium eluate is concentrated to 2.5<u>M</u> Na_2CO_3 + Cs_2CO_3 solution.
- In the steam stripper NH_4OH and $(NH_4)_2CO_3$ are decomposed and NaOH/CsOH in the eluate is converted to Na_2CO_3/Cs_2CO_3. The following reactions occur during the concentration step:

$$NH_4OH(aq) \quad H_2O(\ell) + NH_3(g), \text{ and}$$

$$(NH_4)_2CO_3(aq) \quad H_2O(\ell) + 2NH_3(g) + CO_2(g).$$

- The scrubber bottoms contain 2M $(NH_4)_2CO_3$.
- Essentially all (99.5%) of the NH_3 in the feed to the cesium concentrator is recovered.
- The steam stripper is assumed to operate at 100% efficiency.

Status of Technology for Cesium Eluate Concentration

Experiments with the elutriant recovery/eluate concentration have been performed at Hanford (Richardson 1969). Two operating problems, foaming in the concentrator and $(NH_4)_2CO_3$ pluggage in the condenser, have been resolved through control of equipment operation and equipment design. Foaming in the concentrator was effectively prevented by steam-stripping the feed in

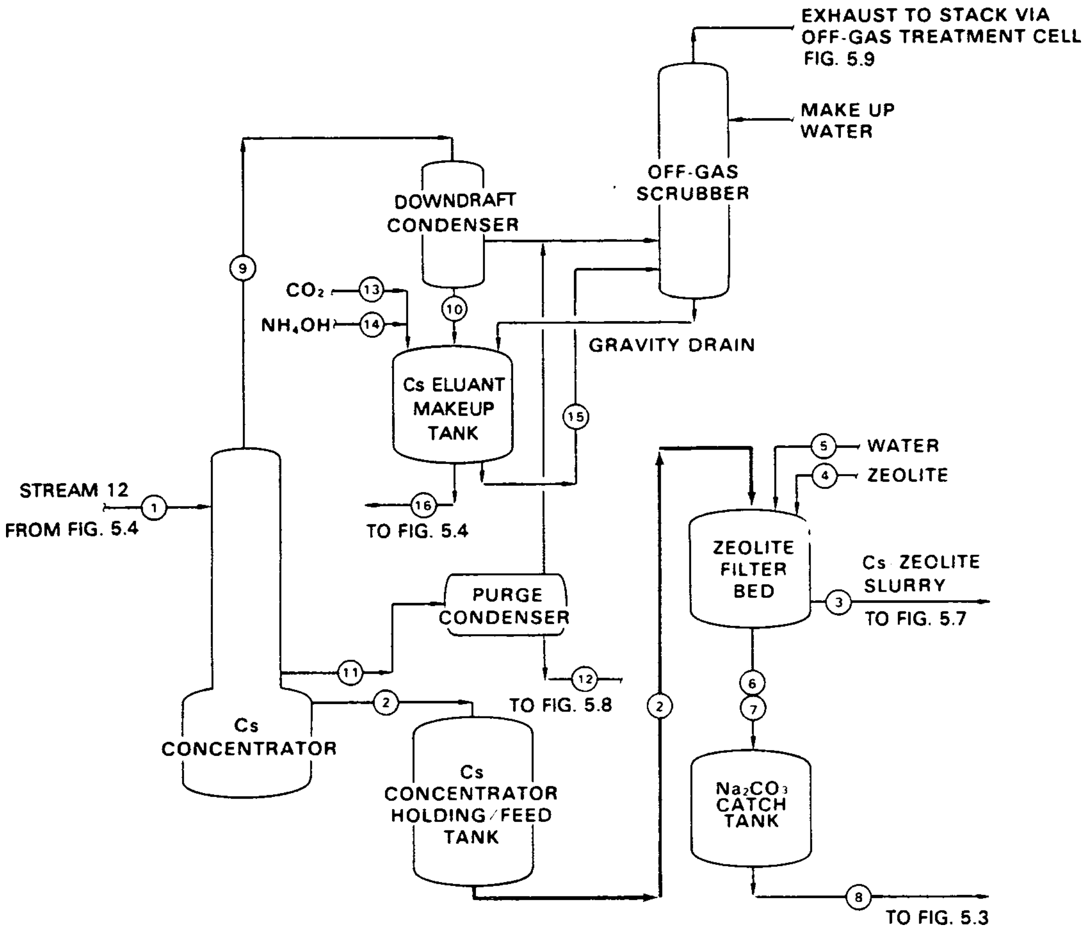

FIGURE 5.5. Flowsheet for Cesium Eluate Concentration and Cesium Fixation on Zeolite

TABLE 5.8. Flowsheet Conditions and Material Balance Data for Cesium Eluate Concentration and Cesium Zeolite Fixation

Operation	Cesium Concentration and Zeolite Fixation			Zeolite Bed Renewal and Washing				
Stream Number	1	2	3	4	5	6	7	8
Description / Variable	Cesium Concentrator Feed	Cesium Concentrator Bottom	Cesium/ Zeolite Slurry	Fresh Zeolite	Zeolite Flush	Zeolite Bed Rinse	Na_2CO_3 Eluate	Recycle Na_2CO_3 Solution
Flow, kg/h								
Na_2CO_3	4.66	4.66				2.33	2.33	4.66
NH_4OH	9.6	--						
$(NH_4)_2CO_3$	15.4	--						
H_2O	117.5	11.7	2.52	1.64	8.97	6.77	13.9	20.67
Zeolite			.33	.33				
CO_2								
Total	147.16	16.36	2.85	1.97	8.97	9.10	16.23	25.33
Temperature, °C	25	102/25	25	25	25	25	25	25
Density, g/cm^3	1.25	1.39	1.13	1.20	1.0	1.34	1.17	1.23
Flow, L/h	117.5	11.7	2.52	1.64	8.97	6.77	13.9	20.6
Activity, Ci/L	10.0	100	465	--	--	--	--	--
Heat, W/L	0.03	0.30	1.4	--	--	--	--	--

TABLE 5.8. (Contd)

Operation	Cesium Eluate Concentration and Cesium Zeolite Fixation						
Stream Number	9	10	11	12	13	14	15
Description / Variable	Cesium Concentrator Overheads	Overheads Condensate	Concentrator Purge	Purge Condensate	CO_2 Makeup Gas	NH_4OH Makeup	Scrub Water
Flow, kg/h							
Na_2CO_3							
NH_4OH	9.6	9.6				2.6×10^{-3}	
$(NH_4)_2CO_3$	15.4	15.4					
H_2O	76.94	76.94	28.86	28.86		1.8×10^{-2}	11.8
Zeolite							
CO_2					1.0		
Total	101.94	101.94	28.86	28.86	1.0	.021	11.8
Temperature, °C	100	35	35	35	35	25	35
Density, g/cm	--	1.32	1.0	1.0	--	1.14	1.0
Flow, L/h	--	77	--	28.86	--	0.018	11.8
Activity, Ci	--	--	--	--	--	--	--
Heat, W/L	--	--	--	--	--	--	--

the packed concentrator tower. This stripping operation removed 99% of the NH_3 and CO_2 from the liquid entering the evaporator. Downdraft condensation prevented pluggage in the condenser. Recoveries of 98% for ammonia and carbon dioxide were demonstrated.

Process Flowsheet for Zeolite Adsorption

Cesium concentrator bottoms consist of a cesium-enriched sodium carbonate solution. To reduce the sodium content the concentrate solution is passed through a fixed zeolite bed which selectively absorbs the cesium from the solution. After loading, the zeolite bed is rinsed with water to remove excess and entrained sodium carbonate. The loaded zeolite bed is then fluidized and transferred to the spray calciner feed tank.

Material Balance for Zeolite Adsorption

The bases used to develop the material balance for cesium adsorption on the zeolite columns are listed below. Table 5.8 gives the material balance data.

- The cesium decontamination factor in the zeolite bed is 100. Cesium not recovered is recycled along with the sodium carbonate solution to the gravity settler feed tank.
- The zeolite is loaded and washed using downflow.
- Loaded zeolite beds are removed with a water slurry with a total volume six times the bed volume.
- The zeolite bed is discharged to the spray calciner feed tank weekly.

Status of Technology for Zeolite Adsorption

Cesium and sodium were first loaded on zeolite in fission-product separation processes at Hanford (Buckingham 1967; Carlstrom 1977). After loading, the cesium and sodium were eluted using a $(NH_4)_2CO_3$-NH_4OH solution. This technique was used to reduce the Na/Cs ratio and to improve ion-exchange efficiencies.

Studies at Savannah River Laboratory (Wiley 1976; Wiley and Wallace 1975) have also been conducted using both tracer-level synthetic ion-exchange eluate and adjusted composition plant wastes. In these tests the feasibility of fixing cesium on zeolite was demonstrated adequately. Approximately 70 column volumes of concentrated column eluate are loaded on one column volume of zeolite before breakthrough. After the initial water wash, the cesium does not leach appreciably from the zeolite.

5.3.4 Strontium Ion Exchange (System D)

The purpose of strontium ion exchange is to remove soluble strontium and any actinides that may remain in the effluent from the cesium ion-exchange column by ion exchange on a Chelex-100 ion exchanger. An additional ion-exchange column is also provided to remove any additional actinides that may still remain in the effluent from the strontium ion-exchange column.

Process Flowsheet

The flowsheet for the strontium ion-exchange process is shown in Figure 5.6. Clarified supernate from the cesium ion-exchange process is pumped through two ion-exchange columns to remove soluble strontium and any actinides that remain from the cesium ion-exchange columns. The first column uses a CHELEX-100 cation exchanger. Decontamination factors of 10^3 have been reported for strontium removal in high pH supernates passed through this resin (Wiley 1976). The second column is a cleanup column designed to remove any residual actinide activity. This column has a nominal decontamination factor of 10 for actinide removal. The specific resin to be used in this column has not been identified.

Effluent from the strontium ion-exchange process is sampled to measure residual radioactivity and is then transferred to the decontaminated supernate solidification process. If the supernate contains fission products and actinides at concentrations that are higher than desired, it is recycled to the strontium ion-exchange feed tank.

The columns are back-flushed with water after loading. Strontium and other absorbed elements are eluted from the column using upflow with $1\underline{M}$ HNO_3. After elution, the column is regenerated with $2\underline{M}$ NaOH. The water rinse from

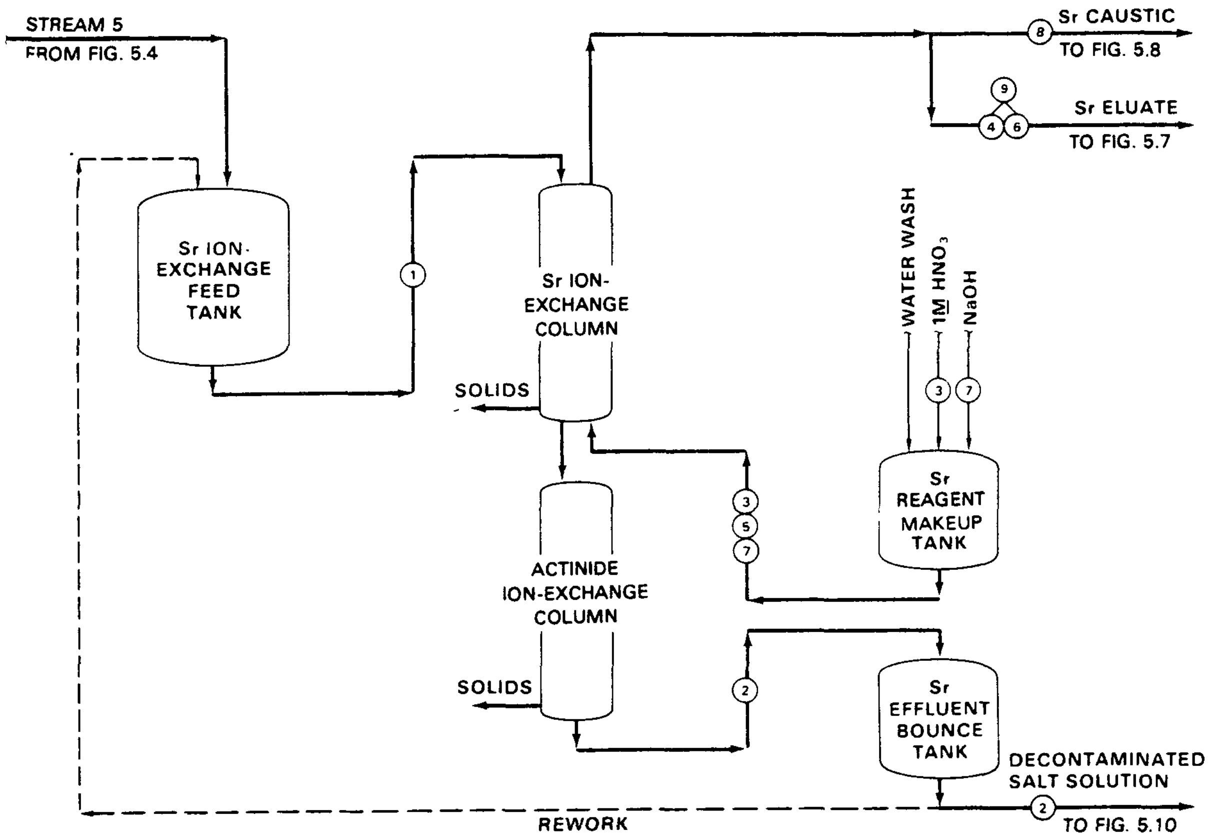

FIGURE 5.6. Flowsheet for the Strontium Ion-Exchange Process

the first column and the strontium eluate solution are transferred to the spray calciner feed tank. Excess caustic from column regeneration is transferred to the recycle evaporator feed tank.

The actinide ion-exchange column is envisioned to be an inorganic ion exchanger resistant to degradation in high pH solutions. This ion exchanger would not be regenerated, but would be disposed of in the spray calciner feed or would be considered transuranic waste and disposed of appropriately.

Process Flowsheet and Material Balance

The bases for the process flowsheet and material balance for strontium ion exchange are presented below. The flowsheet and material balance data are summarized in Table 5.9. The ion-exchange system consists of an exchanger packed with Chelex-100 resin with 50% voidage and no column freeboard and a second exchanger packed with an unidentified ion-exchange resin.

TABLE 5.9. Flowsheet Conditions and Material Balance Data for Strontium Ion Exchange

Operation	Strontium Column Loading, 20 h/batch		Strontium Column Elution, 10 h/batch		Strontium Column Rinse, 1.5 h/batch		Strontium Column Regeneration, 2.0 h/batch		
Stream Number	1	2	3	4	5	6	7	8	9
Description / Variable	Strontium Ion-Exchange Column Feed	Decontaminated Salt Solution	Strontium Acid Eluant	Strontium Eluate	Column Rinse	Rinse Eluate	Caustic Regenerant	Caustic Eluate	Total Eluate to Spray Calciner Feed Tank
Flow, kg/h (kg)[(a)]									
Salts	(1789.2)	(1789.2)							
NaOH							(2.0)	(1.52)	
$NaNO_3$				(5)					0.5
HNO_3			(33)	(28)		(0.3)			3.0
H_2O	(5874.0)	(5874.0)	(506)	(506)	(13.2)	(12.9)	(23.4)	(23.4)	59.2
Na_3CO_3									
Solids	(0.008)	(0.008)							
Total	(7664)	(7664)	(539)	(539)	(13.2)	(13.2)	(25.4)	(24.92)	62.7
Temperature, °C	25	25	25	25	25	25	25	25	25
Density, g/cm^3	1.3	1.3	1.06	1.06	1.0	1.02	1.08	1.06	1.06
Flow, L/h (L)	(5895)	(5895)	(508.5)	(508.5)	(13.2)	(13.2)	(23.5)	(23.5)	59.2
Activity, Ci/L (Ci)	(94.1)	(4×10^{-3})	--	(47.1)	--	--	--	--	0.08
Heat, W/L	0	0	--	0	--	--	--	--	0

(a) Quantities in parentheses indicate batch flow; quantities not in parentheses indicate continuous flow rate.

- The decontamination factor for strontium is 10^3.
- The assumed resin stoichiometries are:

  loading--$2RNa + Sr^{+2}(aq.) = R_2Sr + 2Na^{+}(aq.)$
  elution--$R_2Sr + 2H^{+}(aq.) = 2RH + Sr^{+2}(aq.)$
  regeneration--$RH + Na^{+}(aq.) + OH^{-}(aq.) = RNa + H_2O$.
- The loading step is downflow through the resin bed; other steps are upflow.
- The material balance rates are time-averaged over the cycle time of the ion-exchange bed.
- The time cycle and flow rates are:

Stream	Time, h	Flow Rate, Column Vol/h
Load	20	2
Elute	10	1
Water wash	1.5	1
Regeneration	2	1
Standby	6.5	--
Total	40	--

- A nominal decontamination factor of 10 for actinides is assumed across both the Chelex-100 column and the backup ion exchanger.

Status of Technology

A limited amount of experience has been gained using CHELEX-100 ion-exchange resin for strontium removal in high-caustic supernates. These tests have been limited to laboratory-scale scoping studies. Decontamination factors for strontium of 10^3 have been demonstrated (Wiley 1976a). Removal of plutonium and other actinides from the supernates will occur to some extent. However, no studies directed towards quantifying these decontamination factors have been performed.

The CHELEX-100 resin is a small-beaded resin that packs closely. Feed flow through the column is characterized by a high pressure drop. Smaller columns, such as those proposed for the WNYNSC flowsheet, should be easily operable.

5.3.5 Calcination, Vitrification, and Canister Handling (System E)

The purpose of the calcination/vitrification process is to convert the alkaline high-level waste slurry resulting from waste processing into a borosilicate glass matrix. After filling, the high-level waste canister is cooled and then sealed by welding on a lid. The surface of the canister is decontaminated, and the canister is then removed from the process cell and placed in an interim-storage facility.

Process Flowsheet

A simplified process flowsheet for the calcination/vitrification process is shown in Figure 5.7. Several waste treatment process streams make up the blended alkaline feed stream to be solidified. This blended feed mixture contains sand filter solids, washed sludge, zeolite loaded with cesium, acidic strontium eluate, and other solids from spent off-gas sorber beds comprised of silica gel and silver mordenite.

The blended feed mixture is sampled in one of two spray calciner feed tanks. The sample analyses provide information needed to select the optimum glass-former chemical mixture.

Alkaline high-level waste blend is pumped to the spray calciner using a submerged cantilever pump. The liquid waste is atomized with air in a commercially available spray nozzle into small droplets ($\sim$70 μm) that dry and are decomposed to metallic oxides, called calcine, as they fall through the heated-wall spray calciner chamber. The temperature of the spray chamber furnace varies from 700°C to 900°C depending upon the waste characteristics. The calcine formed is stoichiometrically mixed with the appropriate glass-former chemical mixture as it falls from the spray calciner into one of two canisters located inside an in-can melting furnace. The calcine/glass-former mixture is melted within the heated canisters to form a monolith at a temperature of approximately 1050°C. Once a canister is filled, the calcine/glass-former chemical mixture is diverted to the adjacent canister and the canister filling process is repeated.

The filled glass canisters undergo a fining process. In this process the melting of the calcine and glass-former mixture is completed by holding

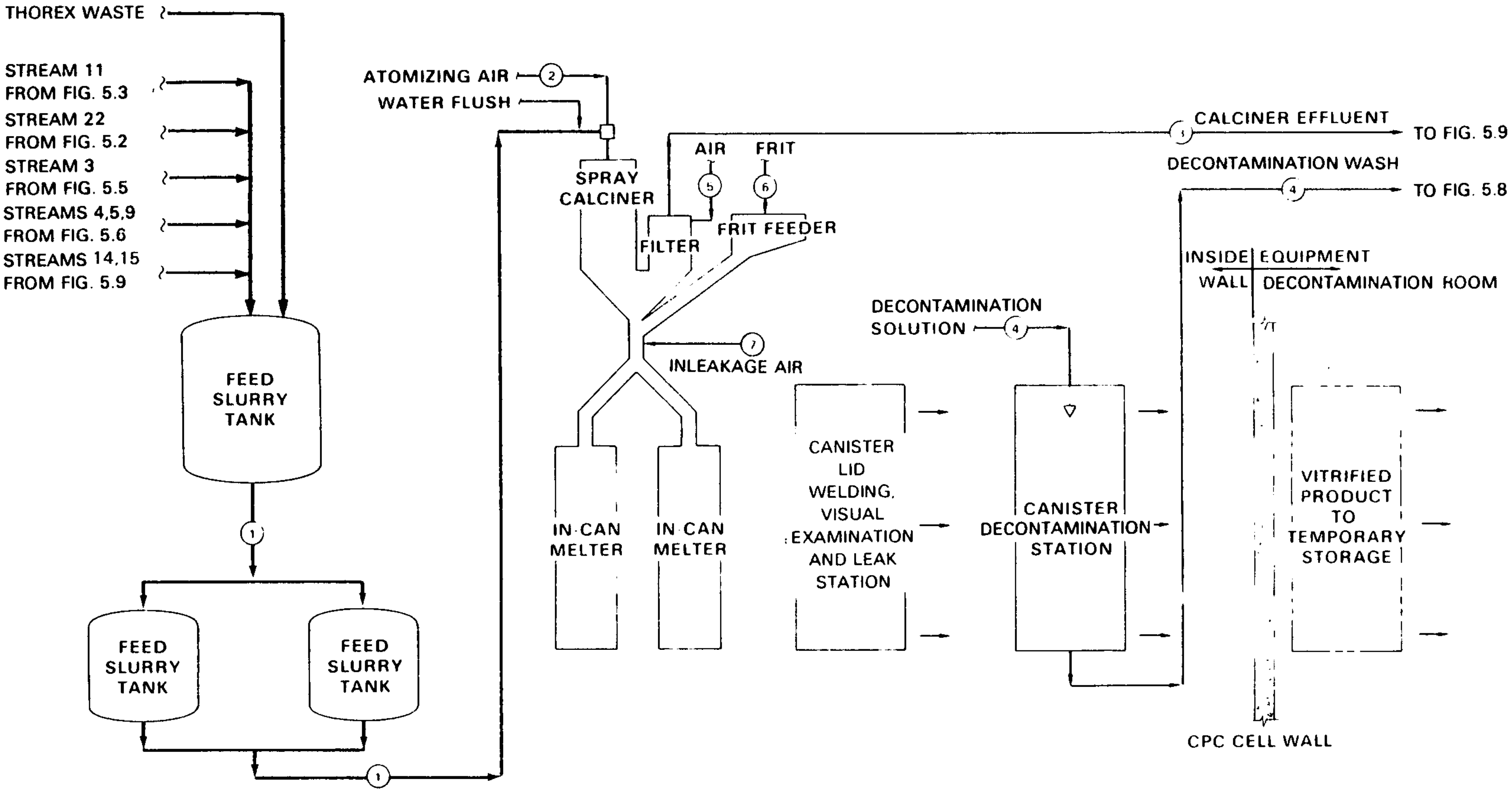

FIGURE 5.7. Flowsheet for the Calcination/Vitrification Process

the temperature at approximately 1050°C until the temperature is uniform throughout the canister. A canister cooldown period is started at the completion of the fining period. The canister is decoupled from the calciner/in-can melter connecting section and is pivoted aside. The tensile strength of the canister is sufficiently high at a temperature of 700°C so that removal of the canister from the in-can melting furnace is possible.

The hot canister is then transferred to a canister lid-welding station and is allowed to cool for several days. A twist-lock lid, which has a helium sponge attached underneath, is placed on the canister and tightened. A remote arc-welding apparatus welds the twist-lock lid/canister interface. The canister lid is then heated, and helium from the sponge is allowed to escape to the air space above the glass monolith. A bell jar is placed on the top of the canister, and the seal weld is helium leak-checked. A leaking weld is repaired by repeating the seal weld process.

Sealed canisters are decontaminated using a standard steam spray. This spray removes canister spall and smearable contamination. The decontaminated canister is then placed upon a canister transfer cart that holds six canisters. The cart transfers the canisters from the Chemical Processing Cell to the Equipment Decontamination Room, where the canisters are further decontaminated before they are transferred to an interim-storage facility for high-level radioactive waste.

Material Balance

The bases used to define the process flowsheet and material balance are summarized below. The flowsheet material balance data are summarized in Table 5.10.

- The atomizing air flow rate is .48 kg of air per L of feed.
- The flow rate of filter blowback air is .60 kg/h.
- All metallic nitrates and hydroxides are assumed to be converted to an oxide during calcination/vitrification operations.
- Operation of the calciner is continuous.
- The weight of the steam used to decontaminate a canister is 20 kg.

TABLE 5.10. Flowsheet Conditions and Material Balance Data for Spray Calcination and Vitrification of Alkaline Waste Slurry

Stream Number	1	2	3	4	5	6	7
Description / Variable	Calciner Feed Slurry	Atomizing Air	Calciner Effluent	Decontamination Wash	Filter Blow-back Air	Glass-Forming Frit	Inleakage Air
Flow, kg/h (kg)(a)							
Salt	3.56						
Solids	7.91					16.7	
H_2O	101.8		101.8	1.0			
Air		49	71.5		0.52		22
NO_2			2.34				
Total	112.27	49	175.64	1.0	0.52	16.7	22
Temperature, ^oC	35	35	300	25	35	35	60
Density, g/cm^3	1.1	--	--	1.0	--	--	--
Flow, L/h	100.8	--	--	1.0	--	--	--
Activity, Ci/L (Ci)	21.6	--	(2.17)	--	--	--	--
Heat, W/L	0.08	--	--	--	--	--	--

(a) Quantities in parentheses indicate batch flow; quantities not in parentheses indicate continuous flow rate.

- The composition of the alkaline high-level waste feed is summarized in Table 5.11. This feed composition is an average composition based on continuous flow of all process feed streams.
- The glass-former chemical mixture is summarized in Table 5.12. This composition was estimated based upon the nuclear waste composition given in Table 5.11.
- The waste oxide loading in the vitreous product is 25 wt%.

Status of Technology

The spray calciner/in-can melter process is the most highly developed of the waste solidification processes being considered in the United States at this time. The generic process is essentially ready for plant application. However, some site-specific development work may be required to implement the process. The spray calciner/in-can melter process involves many mechanical operations that must be performed remotely on a routine basis in a production facility. Some of these process steps have not been performed on a remote production scale; however, the specific details of these operations have been developed and demonstrated in a nonradioactive environment (Larson 1980).

5.3.6 Recycle and Secondary Evaporation (System F)

The purpose of recycle and secondary evaporation is to dewater process waste streams and return essentially all of the radionuclide contamination to the immobilization process. Decontaminated evaporator overheads are condensed and transferred to the low-level waste treatment facility.

Process Flowsheet

Figure 5.8 is a simplified process flowsheet of the recycle and secondary evaporator process. This system consists of the recycle feed blend and holding tank, two thermosyphon evaporators equipped with downdraft condensers, a recycle evaporator bottoms holding tank, a recycle water holding tank and a condensate sample tank.

Feed to the recycle evaporator consists of a blended mixture of several process waste streams. These process streams are secondary evaporator bottoms,

TABLE 5.11. Estimated Chemical Composition of Salt/Sludge Separated Alkaline Waste Blend Fed to Solidification Process

Component	Concentration, g/L
Solids	
$Fe(OH)_3$	33.9
$FePO_4$	19.5
$Cr(OH)_3$	2.6
$Ni(OH)_2$	1.2
$Al(OH)_3$	.6
$RE(OH)_3$	1.1
$FP(SO_4)$	.9
$FP(OH)_4$	1.1
FP_2O_3	.1
Na_3PO_4 $12MoO_3$	.7
MnO_2	1.5
PuO_2	.02
$Na_2U_2O_7$	4.6
Actinide Oxide	.009
Other solids(a)	9.1
Total	76
Soluble Salts	
$NaNO_3/NO_2$	5.4
Na_2SO_4	.04
NaOH	.008
NaCl	.0003
$FePO_4$	.006
$Cr(OH)_3$	.001
$Ni(OH)_2$	.0004
AlF_3	.0008
$Na_3PO_4 \cdot 12MoO_3$	.0003
Total	5.5
Water	102 (L/h)

(a) .06 Ag mordenite, 3.24 zeolite, .19 silica gel, and 5.59 SiO_2.

TABLE 5.12. Representative Glass-Former Mixture for Vitrification of Alkaline Waste Sludge Slurry[(a)]

Component	Quantity, wt%
SiO_2	49.3
B_2O_3	16.4
Li_2O	2.7
Na_2O	15.1
K_2O	1.4
CaO	2.7
TiO_2	2.7
MgO	2.7
BaO	2.7
ZrO_2	2.7

(a) Based on waste composition shown in Table 5.11.

liquids from the process cell sumps, liquid waste from the analytical support cells, decontamination solutions, wash centrate from the sludge washing module, caustic rinse solutions from the cesium and strontium ion-exchange system, and condensate purged from the cesium concentrator.

Bottoms from the recycle evaporator are collected in a holding tank. The recycle evaporator bottoms contain all of the salt and solids that enter in the evaporator feed. These bottoms are periodically transferred to the gravity settler feed tank. Overheads from the recycle evaporator comprise the feed to the secondary evaporator. In the secondary evaporator essentially all of the remaining contamination is removed from the process condensates. The bottoms are returned to the recycle evaporator. Overheads from the secondary condenser are condensed and collected in a recycle water tank. To the extent possible, process condensates are recycled and used again in the process. A sample tank adjoins the condensate hold tank. Process condensates are sampled and if they are found to be below permissible contamination limits they may be released to the low-level waste treatment facility.

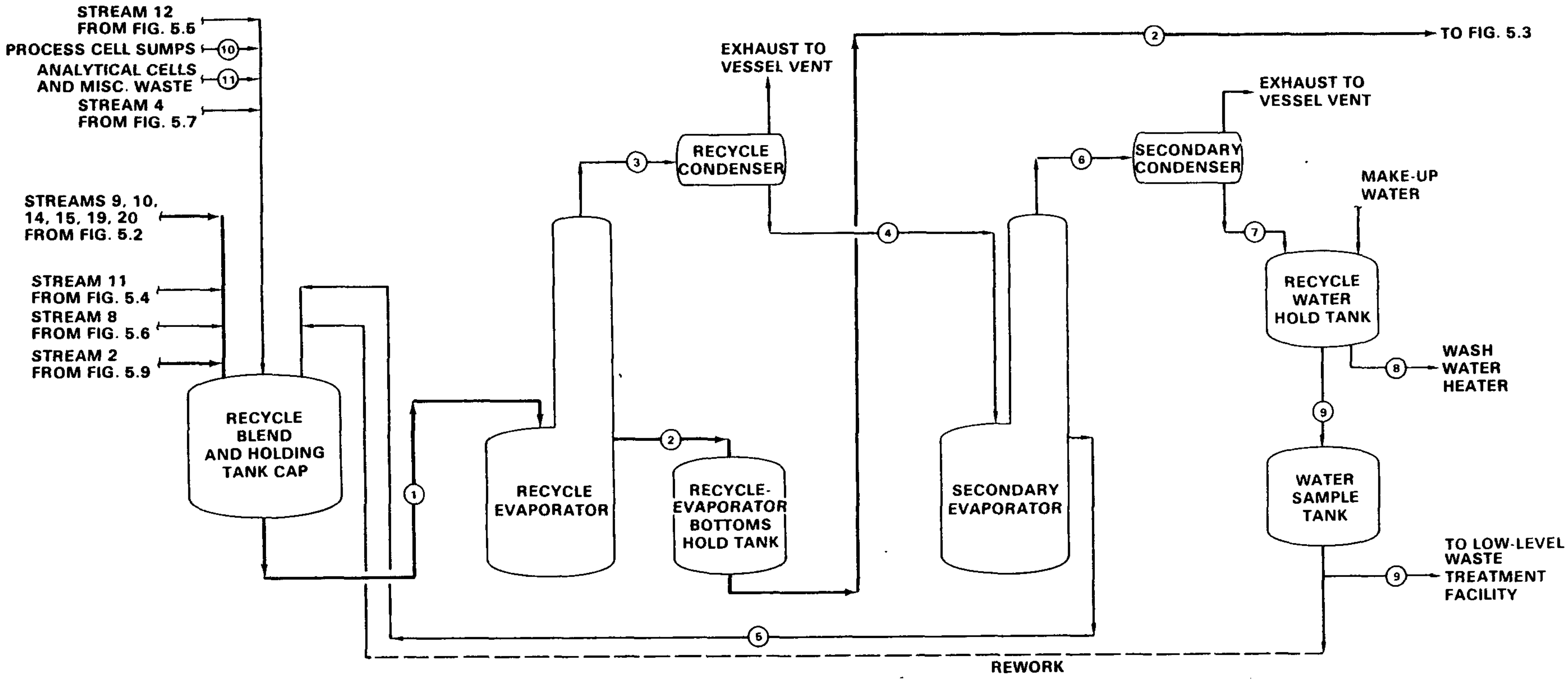

FIGURE 5.8. Flowsheet for Recycle and Secondary Evaporation

The bases used in developing the material balance for the recycle and secondary evaporation process are summarized below. The material balance data are presented in Table 5.13.

- A decontamination factor of 10^6 for radiochemical activity is assumed across both the recycle evaporator and the secondary evaporator.
- The bottoms from the recycle evaporator are concentrated to 33 wt% salt.
- An overheads-to-bottoms split of 10 to 1 is assumed in the secondary evaporator.

5.3.7 Calciner Off-Gas Treatment (System G)

The purpose of the calciner off-gas treatment system is to remove radionuclide and chemical contaminants from the calciner effluents before the gases are released to the environment.

Process Description

A flow diagram of the calciner effluent treatment system is shown in Figure 5.9. The unit operations to remove radionuclide and chemical contamination from the calciner effluents are wet scrubbing, de-entrainment, condensation, sorption, filtration, and chemical destruction. The equipment to effect these operations are a venturi scrubber/cyclone, a condenser, a demister, heaters, packed sorption beds, filters, and an NO_x destructor.

Off gases from the spray calciner/in-can melter process consist primarily of water vapor, NO_x from nitrate decomposition, and possibly volatile forms of sulfur, chloride, and fluoride. Volatile radionuclides include cesium, strontium and antimony.

Initial decontamination of the calciner effluents is provided by the spray calciner filters. These filters remove particulates, allowing volatile chemical and radionuclide species to pass through. The venturi scrubber/cyclone unit provides further particulate decontamination and condenses a fraction of the water vapor in the effluents. The venturi scrubber also provides the necessary safety margin should the calciner filters be breached.

TABLE 5.13. Flowsheet Conditions and Material Balance Data for Recycle and Secondary Evaporation

Stream Number	1	2	3	4	5
Description / Variable	Recycle Evaporator Feed	Recycle Evaporator Bottoms	Recycle Evaporator Overheads	Secondary Evaporator Feed	Secondary Evaporator Bottoms
Flow, kg/h					
Salt	20.84	20.84			
Solid	.604	0.604			
H_2O	647.6	59.4	588.2	588.2	53.5
Total	669.04	80.84	588.2	588.2	53.5
Temperature, ^0C	35	100/60	100	100	100
Density, g/cm^3	1.03	1.36	--	1.0	1.0
Flow, L/h	648	59.4	--	588.2	53.5
Activity, Ci/L	0.52	5.67	--	1×10^{-6}	1×10^{-6}
Heat, W/L	0.001	0.10	--	0	0

TABLE 5.13. (Contd)

Stream Number	6	7	8	9	10	11
Description / Variable	Secondary Evaporator Overheads	Condensate from Secondary Condenser	Wash Water	Wash Water	Process Cell Sumps	Analytical Cell and Miscellaneous Wastes
Flow, kg/h						
Salt						
Solid						
H_2O	534.7	534.7	534.7		25	25
Total	534.7	534.7	534.7		25	25
Temperature, °C	100	25	25	25	35	35
Density, g/cm^3	--	1.0	1.0	1.0	1.0	1.0
Flow, L/h	--	534.7	534.7		25	25
Activity, Ci/L	--	1×10^{-12}	1×10^{-12}	--	--	
Heat, W/L	--	0	0		--	--

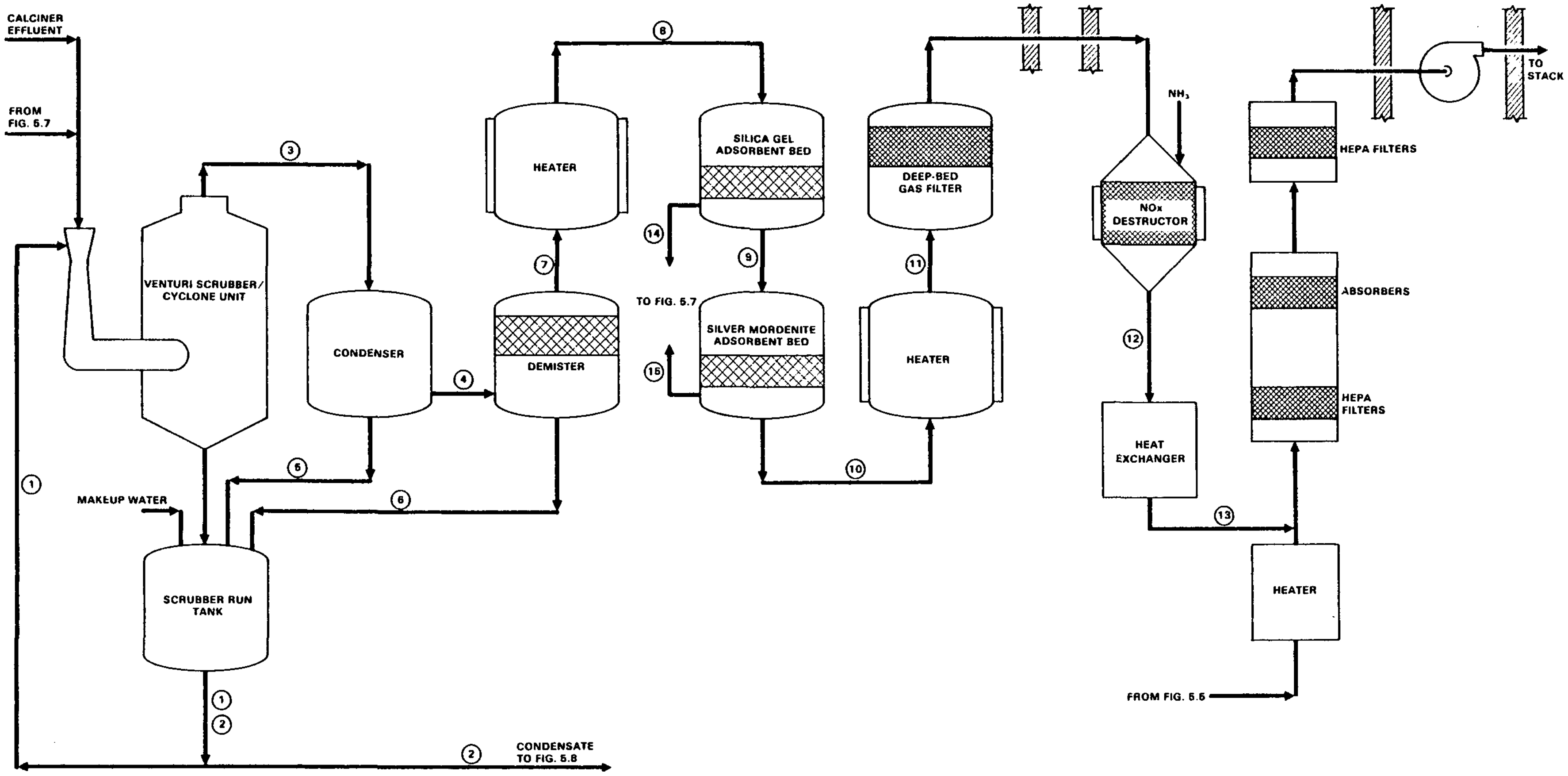

FIGURE 5.9. Flowsheet for Calciner Off-Gas Treatment

The condenser and demister remove essentially all of the water vapor present in the gaseous effluent. The majority of the radionuclide decontamination also occurs in the condensation process.

Dry scrubbing processes to remove volatile radionuclides occur in the silver mordenite bed, which removes iodine, and the silica gel bed, which removes ruthenium and possibly cesium and antimony. The gas stream is heated by in-line heaters to assure that no condensation occurs in the sorption beds and that more efficient sorption occurs. A deep-bed fiberglass gas filter that removes fine particulate is located in the off-gas cleanup system immediately before the off-gas exits from the Chemical Processing Cell.

The high-efficiency particulate air filters located in the off-gas cleanup cell provide additional removal of fine particulates. These filters also serve as an additional safety factor for decontamination between the process cell and the environment should a process system accident occur.

The NO_x formed from the decomposition of the metallic nitrates in the calcination process is not easily removed by normal scrubbing, filtration, or sorption processes. Therefore, a chemical reactor is used. This reactor operates by catalytically reacting ammonia and NO_x to form water and nitrogen. Sufficient NO_x destruction is provided so that the effluent stream may be released to the environment through the building stack.

Material Balance

The bases for developing the material balance for the calciner effluent treatment system are summarized below. The material balance data are summarized in Table 5.14.

- All radionuclide contaminates are assumed to behave as particulates.
- Seventy-five percent of the water vapor in the calciner off gas is condensed in the venturi scrubber; the remaining water vapor in the off gas is removed in the downdraft condenser and the demister.
- The decontamination factor for particulates in the spray calciner/ in-can melter system off gas is 10^3.

TABLE 5.14. Flowsheet Conditions and Material Balance Data for Calciner Off-Gas Treatment

Stream Number	1	2	3	4	5	6	7	8
Description / Variable	Off-Gas Venturi Scrub Solution	Condensate to Recycle Blend Tank	Cyclone Off Gas	Off Gas to Demister	Condensate to Scrub Tank	Condensate to Scrub Tank	Off Gas to Heater	Off Gas to Silica Gel Bed
Flow, kg/h								
H_2O	9606	101.8	25.4	3.0	22.4	3.0		
Air			71.5	71.5			71.5	71.5
NO_x			2.34	2.34			2.34	2.34
Solids								
Total	9606	101.8	99.24	76.84	22.4	3.0	73.84	73.84
Temperature, °C	60	60	70	20	20	20	20	8
Density, g/cm^3	1.0	1.0	--	--	1.0	1.0	--	--
Flow, L/h	9606	101.8	--	--	22.4	3.0	--	--
Activity, Ci/L, (Ci)	2.3×10^{-4}	0.021	(2.2×10^{-2})	(2.2×10^{-4})	2×10^{-6}	2×10^{-6}	(2.2×10^{-4})	(2.2×10^{-4})
Heat, W/L	0	--	--	--	0	0	--	--

TABLE 5.14. (contd)

Stream Number	9	10	11	12	13	14	15
Description / Variable	Off Gas to Silver Mordenite Bed	Off Gas to Heater	Off Gas to NO_x Destructor	Off Gas to Heat Exchanger	Off Gas to Atmospheric Protection System	Silica Gel to Feed Slurry Hold Tank[a]	Silver Mordenite to Feed Slurry Hold Tank[a]
Flow, kg/h							
H_2O							
Air	71.5	71.5	71.5	71.5	71.5		
NO_x	2.34	2.34	2.34	0.0234	0.0234		
Solids						0.02	0.006
Total	73.84	73.84	73.84	71.52[b]	71.52	0.02	0.006
Temperature, °C	80	80	80	350	80	35	35
Density, g/cm^3	--	--	--	--	--	--	--
Flow, L/h	--	--	--	--	--	--	--
Activity, Ci/L (Ci)	(2.2×10^{-6})	(2.2×10^{-6})	(2.2×10^{-7})	(2.2×10^{-7})	(2×10^{-7})	--	--
Heat, W/L	--	--	--	--	--	--	--

(a) Batch operation.

(b) Original air only. A small amount of N_2 and H_2O is added from the catalytic reduction of NO_x with NH_3.

- The decontamination factors for the off-gas equipment are summarized below:

Equipment	Decontamination Factor[(a)] Particulate	NOx
Venturi scrubber	10^2	1
Cyclone	1	1
Condenser	10^2	1
Demister	1	1
Silica gel sorber	10^2	1
Ag mordenite sorber	1	1
Fiberglass filter	10	1
NO_x destructor	1	10^2
Sintered filters (calciner/melter system)	10^3	1

(a) Adapted from Christian and Pence (1977).

Status of Technology

The calciner effluent treatment system is well characterized and ready for plant application. A large quantity of technical performance data have been developed for nuclear waste solidification off-gas cleanup systems. Effluent source term data for various waste types have been developed for the spray calciner/in-can melter system. A thorough review of design data, radioactive source terms and effluent decontamination characteristics can be found in Christian et al. (1978), Christian and Pence (1977), Christian and Rhodes (1977), Hanson (1980), Rimshaw, Case and Tompkins (1980), Thomas and Munger (1978), and Larson (1980).

5.3.8 Solidification of Decontaminated Supernate (System H)

The purpose of the decontaminated supernate solidification process is to dewater the supernate to form a salt cake containing 20 to 25 wt% water. The salt cake product is packed in 55-gal drums for interim storage.

Process Flowsheet

A schematic flowsheet for solidification of decontaminated supernate is shown in Figure 5.10. Decontaminated salt solution from the strontium ion-exchange process is dewatered in a two-stage evaporation process. Decontaminated salt solution is fed to a primary evaporator in which the water content of the salt solution is reduced from 77 to 40 wt%. The resulting concentrated salt solution is fed to a horizontally mounted wiped-film evaporator. The wiped-film evaporator further dewaters the salt solution until a product with 20 to 25 wt% water is produced. This salt cake product is packaged and allowed to cool in 55-gal drums. Upon cooling, the concentrate forms a damp crystallized salt. These drums are transferred to interim storage.

Condensate is collected from both the primary supernate evaporator and the wiped-film evaporator. This condensate is sampled to determine activity levels. If required, the condensate is recycled to the primary evaporator's feed tank. Normally the condensate is transferred to the low-level waste treatment system.

Material Balance

The bases used to develop the material balance for the solidification of decontaminated supernate are summarized below. The material balance data are summarized in Table 5.15.

- The salt solution is evaporated to 60 wt% water in the primary evaporator.
- The salt solution is evaporated to 22 wt% water in the wiped-film evaporator.
- A decontamination factor of 10^6 is assumed in both the primary evaporator and the wiped-film evaporator.

Tables 5.16 and 5.17 provide estimates of the chemical composition and the radionuclide composition of the solidified salt cake.

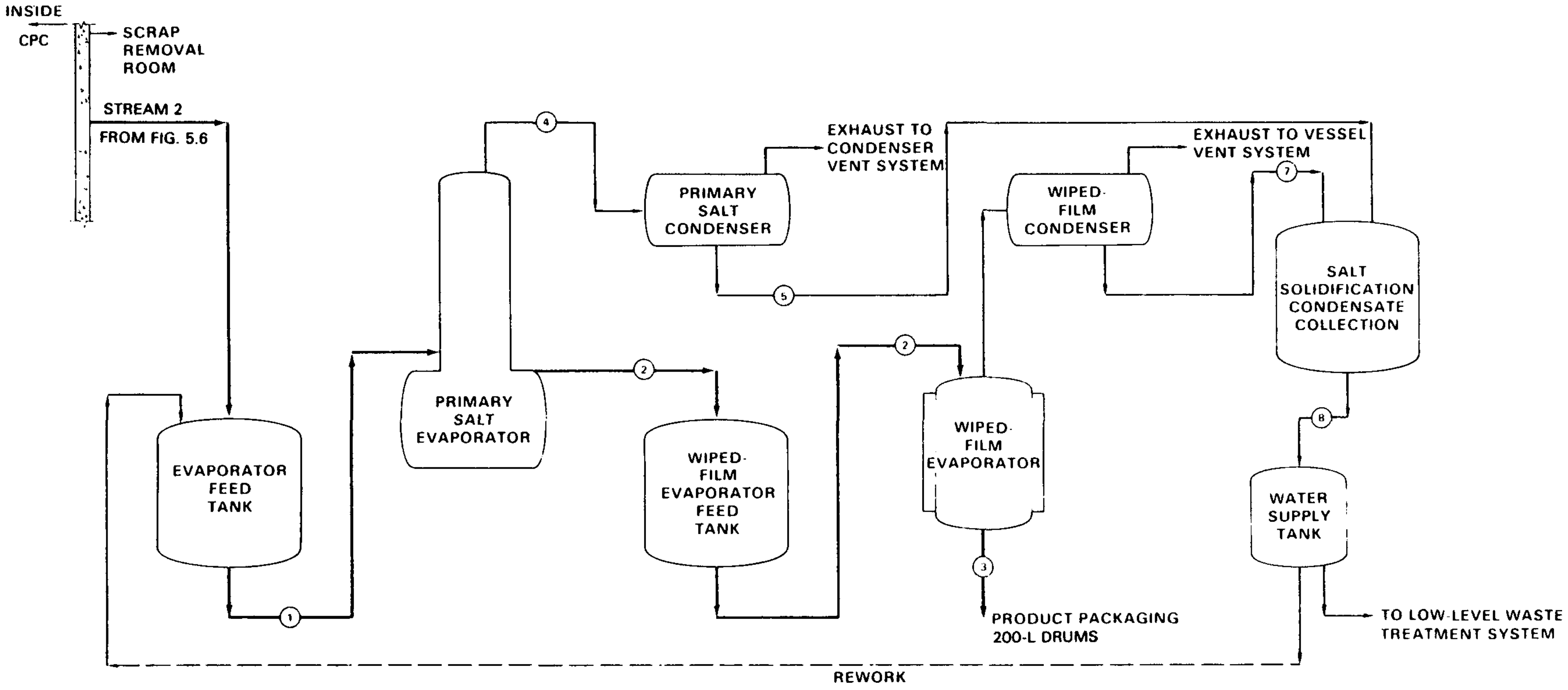

FIGURE 5.10. Flowsheet for Decontaminated Supernate Solidification

TABLE 5.15. Flowsheet Conditions and Material Balance Data for Solidification of Decontaminated Supernate

Stream Number	1	2	3	4
Description / Variable	Evaporator Feed	Wiped-Film Concentrated Salt Solution	Evaporator Product	Evaporator Overheads
Flow, kg/h				
Salt	89.5	89.5	89.5	
H_2O	293.7	134.2	25.2	159.5
Solids	0.0004	0.0004	0.0004	
Total	383.16	223.7	114.7	159.5
Temperature, ^oC	35	105	110	
Density, g/cm^3	1.30	1.67	1.70	
Flow, L/h	294	134.2	67.5	
Activity, Ci/L (Ci)	6×10^{-7}	1.3×10^{-6}	2.6×10^{-6}	--
Heat, W/L	0	0	0	

Stream Number	5	6	7	8
Description / Variable	Evaporator Condensate	Evaporator Overheads	Overhead Condensate	Process Condensate
Flow, kg/h				
Salt				
H_2O	159.5	109	109	268.5
Solids				
Total	159.5	109	109	268.5
Temperature, ^oC	35	100	35	35
Density, g/cm^3	1.0	--	1.0	1.0
Flow, L/h	160	--	109	268.5
Activity, Ci/L (Ci)	1×10^{-12}	--	1×10^{-12}	1×10^{-12}
Heat, W/L	0	--	0	0

TABLE 5.16. Estimated Salt Cake Composition--Salt/Sludge Separation Process

Component	Quantity, wt%
(Na,K)SO4[a]	5.3
(Na,K) NO_3/NO_2[a]	65
(Na,K) OH[a]	2.7
(Na,K) Cl[a]	.03
$FePO_4$	.07
$Cr(OH)_3$	.1
$Ni(OH)_2$	.04
Na_2CO_3[b]	4.6
AlF_3	.09
H_2O[c]	22

(a) Sodium-to-potassium ratio is 99 to 1.
(b) Formed during cesium ion-exchange processes.
(c) Salt concentrated to 22 wt% water.

TABLE 5.17. Calculated Isotopic Composition of Low-Level Salt Cake--Salt/Sludge Separation Process

Isotope	Half-Life	Percent Insoluble(a)	Activity in 1987, Ci	Activity in in Sludge, Ci	Activity in Supernate, Ci	Decontamination Factor for Insoluble Isotopes	Decontamination Factor for Soluble Isotopes	Activity in Salt Cake(b) nCi/gram
Se-79	6.5 x 10^4 y	0	50	0	50	0	0	30.4
Sr-90	29 y	99	6.7 x 10^6	6.6 x 10^6	6.7 x 10^4	1.7 x 10^4	10^3	230
Zr-93	1.5 x 10^6 y	99.9	250	250	0.2	1.7 x 10^4	0	0.16
Nb-93m	13.6 y	99.9	240	240	0.2	1.7 x 10^4	0	0.15
Tc-99	2.13 x 10^5 y	0	1900	0	1900	0	0	1200
Ru-106	368 d	99	110	110	1	1.7 x 10^4	0	0.67
Sb-125	2.71 y	99	6100	6100	60	1.7 x 10^4	0	37
Cs-134	2.06 y	0	21000	0	21000	0	10^4	1.3
Cs-137	30.2 y	0	8.9 x 10^6	0	8.9 x 10^6	0	10^4	540
Pm-147	2.62 y	99.9	61000	61000	60	1.7 x 10^4	0	39
Sm-151	93 y	99.9	200,000	200,000	200	1.7 x 10^4	0	130
Eu-152	13.4 y	99.9	410	410	0.4	1.7 x 10^4	0	.24
Eu-154	8.2 y	99.9	130,000	130,000	130	1.7 x 10^4	0	84
Np-239	2.35 d	99.9	220	220	0.2	1.7 x 10^4	0	0.08
Pu-238	87.7 y	99.9	1500	1500	1.5	1.7 x 10^4	3 x 10^3	0.05
Pu-239	24,370 y	99.9	1800	1800	1.8	1.7 x 10^4	3 x 10^3	0.06
Pu-240	6580 y	99.9	970	970	1.0	1.7 x 10^4	3 x 10^3	0.03
Pu-241	13.2 y	99.9	70,000	70,000	70	1.7 x 10^4	3 x 10^3	2.5
Am-241	458 y	99.9	20,000	20,000	20	1.7 x 10^4	3 x 10^3	0.7
Cm-244	17.6 y	99.9	8800	8800	9	1.7 x 10^4	3 x 10^3	0.3
Total	--	--	--	--	--	--	--	2300

(a) DOE (1978; Table 3.7).
(b) Total salt cake weight is 1.6 x 10^6 kg.
(c) Total actinide activity ∿3.7 nCi/g salt cake.

5.4 DETAILED PROCESS FLOWSHEET--SOLIDIFICATION OF THE ACIDIC THOREX WASTE

Summarized in this section are the process assumptions, process flowsheets, and the chemical and radionuclide material balances for solidification of the acidic Thorex waste. The Thorex waste will be solidified in a separate campaign, which will precede solidification of the alkaline Purex waste.

5.4.1 Purpose of Separate Processing of Thorex Waste

The reasons for separate processing of the Thorex waste are minimization of required process facilities and simplified plant operations. When the acidic Thorex waste is solidified first, its storage tank (Tank 8D4) is made available for use as a possible surge tank to buffer the transfer of homogenized alkaline waste from its storage tank (Tank 8D2) to the Chemical Processing Cell. This surge tank can be equipped with a much smaller-capacity transfer pump (20 to 40 L/min) as compared to the pump that would be installed on Tank 8D2. The waste receiving tank inside the Chemical Processing Cell is small (12,000 L) and would fill quickly with a larger pump flow with the potential for accidental overflow. If Tank 8D4 were equipped with a small transfer pump, the need to use the spare noncontaminated stainless steel tank (8D3) in the tank farm, or to construct a separate surge tank at the tank farm, would be eliminated.

The acidic Thorex and alkaline Purex waste should not be blended in Tank 8D2. This blended mixture could produce an alkaline sludge with physical properties that would make the sludge/supernate separation difficult to achieve with high efficiency. This blending approach would also increase by approximately 25% the weight of the sludge/supernate mixture requiring separation.

The acidic Thorex waste and the processed alkaline waste blend could be blended simultaneously in the Chemical Processing Cell. However, this approach would require additional transfer lines, controls, and surge tanks in the process cell and at the tank farm.

5.4.2 Process Flowsheet

The acidic Thorex waste will be solidified using the same processing equipment used to solidify the alkaline waste blend. The unit operations used from the alkaline waste treatment and solidification flowsheet are:

- the calcination/vitrification and canister handling equipment (see Figure 5.7; except that the only feed will be acidic Thorex waste from Tank 8D4 and recycle evaporator bottoms);
- the vitrification effluent cleanup equipment (see Figure 5.9);
- the recycle and secondary evaporation equipment (see Figure 5.8).

The acidic Thorex waste is solidified without waste pretreatment. It is periodically transferred from Tank 8D4 to the waste receiving tank located inside the Chemical Processing Cell. The waste is then transferred to the feed blend and adjustment tank. Here the waste is sampled. The waste blend is analyzed, and the analytical results are used to choose the proper glass frit blend to be used for vitrification.

After sampling and analysis, the Thorex waste is converted to a borosilicate glass using the spray calciner/in-can melter process. The rate of glass production is limited by the maximum melting rate of the in-can melter furnace. Maximum glass production rates of 80 kg glass/h are assumed in the 2-ft-dia high-level waste canisters (Larson 1980).

5.4.3 Material Balance

The bases used to define material balance for solidification of the acidic Thorex waste are described below.

- The acidic Thorex waste will be solidified in a 60-day campaign.
- The volume of the Thorex waste plus transfer line flushes in 50,000 L.
- The Thorex waste oxide loading in the final glass product is 15 wt%. This corresponds to ~10 wt% thoria (ThO_2) in the glass.
- The acidic Thorex waste composition is defined in Tables 5.3 and 5.4.
- All process operating parameters for the calciner effluent treatment system are the same as those for solidification of the alkaline waste slurry (see Section 5.3.7).

The material balances are summarized in Tables 5.18, 5.19, and 5.20.

TABLE 5.18. Flowsheet Conditions and Material Balance Data for Spray Calcination and Vitrification of Acidic Thorex Waste

Stream Number[(a)]			1	2	3	4	5	6	7	
Description / Variable	Thorex Waste From Tank Farm	Recycle Evaporator Bottoms	Thorex Feed to Spray Calciner	Atomizing Air	Calciner Off Gas	Canister Decontamination Spray	Filter Blowback Air	Frit Addition	Inleakage Air	Glass Production Rate
Flow, kg/h										
Salt	27.9		27.9							
Solids								68		81.3
H_2O	34.7	10.5	45.2		45.2	0.8				
Air				16	26.2		0.2		10	
NO_x					10.9					
Total	62.6	10.5	73.1	16	82.3	0.8	0.2	68	10	81.3
Flow, L/h	35	10.5	45.5							
Temperature, ^{o}C	35	60	35		300	100		35	35	1050
Density, g/cm^3	1.8	1.0	1.7			1.0		2.0 (bulk)		2.8 ($35^{o}C$)
Activity, Ci/L (Ci)	48	0.16	39.1		(1.7)	Negligible				57.8

(a) Identified stream numbers are from Figure 5.7.

TABLE 5.19. Flowsheet Conditions and Material Balance Data for Calciner Off-Gas Treatment

Stream Number(a)		1	2	3	4	5	6
Description / Variable	Spray Calciner Off Gas(b)	Venturi Scrub Solution	Condensate to Recycle Evaporator	Off Gas From Venturi/Scrubber	Off Gas From Condenser	Condensate From Condenser	Condensate From Demister
Flow, kg/h							
H_2O	45.2	9632	42.5	10.5	5.25	5.25	5.25
Air	26.2			26.2	26.2		
NO_x	10.9			10.9	10.9		
Total	79.6		42.5	47.6	42.4	5.25	5.25
Temperature, °C	300	60	60	60	20	20	20
Density, g/cm^3		1.0	1.0	1.0			
Activity, Ci	1.7	1.7	1.7	1.7×10^{-2}	1.7×10^{-4}	10^{-4}	10^{-4}

(a) Stream numbers are identified on Figure 5.9, unless otherwise indicated.
(b) Stream 3 from Table 5.18.

TABLE 5.19. (Contd)

Stream Number[a]	7	8	9	10	11	12	13
Description / Variable	Off Gas From Demister	Off Gas to Silica Gel Bed	Off Gas to Ag Mordenite Bed	Off Gas to Heater	Off Gas to NO_x Destructor	Off Gas to Exchanger	Off Gas to Atmospheric Protection System
Flow, kg/h							
H_2O							
Air	26.2	26.2	26.2	26.2	26.2	26.2	26.2
NO_x	10.9	10.9	10.9	10.9	0.11	0.11	0.11
Total	37.1	37.1	37.1	37.1	26.3[b]	26.3	26.3
Temperature, °C	20	80	80	35	200	350 to 500	35
Density, g/cm^3							
Activity, Ci	10^{-4}	10^{-4}	10^{-6}	10^{-6}	10^{-7}	10^{-7}	10^{-7}

(a) Stream numbers are identified on Figure 5.9, unless otherwise indicated.
(b) Original air only. A small amount of N_2 and H_2O is added from catalytic reduction of NO_x with NH_3.

TABLE 5.20. Flowsheet Conditions and Material Balance Data for Recycle and Secondary Evaporation

Stream Number[(a)]	1	2	3	5	6
Description / Variable	Feed to Recycle Evaporator	Recycle Evaporator Bottoms	Recycle Evaporator Overheads	Secondary Evaporator Overheads	Secondary Evaporator Bottoms
Flow, kg/h					
H_2O	105.5	10.5	95	85.5	9.5
Total	105.5	10.5	95	85.5	9.5
Temperature, °C	45	105	100	100	100
Density, g/cm^3	1.0	1.0	1.0	1.0	1.0
Activity Ci/L (Ci)	0.016	0.16	(10^{-5})	(10^{-9})	(10^{-5})

Feed Stream Description / Variable	Canister Decontamination Spray	Process Cell Sump	Analytical and Miscellaneous Waste	Off-Gas Scrub Solution
Flow, kg/h				
H_2O	0.8	25	25	45.2
Total	0.8	25	25	45.2
Temperature, °C	80	35	35	60
Density, g/cm^3	1.0	1.0	1.0	1.0
Activity Ci/L (Ci)	Negligible	Negligible	Negligible	0.037

(a) Stream numbers are identified on Figure 5.8.

5.4.4 Status of Technology

Thorium is recovered in the normal Thorex fuel reprocessing flowsheet. Waste solidification development efforts have therefore been directed towards immobilization of an acidic waste with a low thorium concentration. Nonradioactive development studies have been conducted on such simulated Thorex wastes using the spray calciner/in-can melter process. Cerium and other rare earth elements were used as a thorium substitute. The purpose of these tests was to evaluate chloride and fluoride corrosion on calciner and in-can melter construction materials (Hill 1978).

Thorium was not recovered during reprocessing of the thoria fuel at the WNYNSC. The Thorex waste is therefore dominated by a high concentration of thorium, which presents unique problems in waste management. Thoria has a limited solubility in nuclear waste glasses. It is estimated that this solubility limit is comparable to other actinide elements. Walker and Riege (1979) indicate that actinide solubilities in high-alkaline borosilicate glasses range from 5 to 10 wt%. Glass development studies at Pacific Northwest Laboratory have shown that precipitation of cerium and thorium from the borosilicate glass matrix will not significantly affect its chemical durability. The same behavior is expected for thorium in the WNYNSC glass.

Thoria crystal precipitation from the borosilicate glass matrix does not represent a processing constraint for the spray calciner/in-can melter process. This is because the vitreous product is formed inside the containment canister. In a liquid-fed ceramic melter, thoria crystals would probably accumulate at the bottom of the melter, forming a slag on the melter floor. This slag could eventually stop flow from the melter. Development efforts would be needed to evaluate this prediction if a liquid-fed melter were to be used for waste solidification.

5.5 WASTE SOLIDIFICATION PROCESS INTERFACES

The waste treatment and solidification system is one part of the waste management facilities required for the West Valley Demonstration Project. Other facilities that support the waste solidification effort are needed. These interfacing facilities, shown in Figure 5.11, are not part of the Pacific Northwest Laboratory/Vitro Engineering Corporation preconceptual design. The major facilities that are required to support the waste solidification process are:

- Waste Retrieval Facility--This facility includes all equipment such as sluicing pumps and agititators, and containment buildings required to transfer the high-level liquid waste from the tank farm to the Chemical Processing Cell.

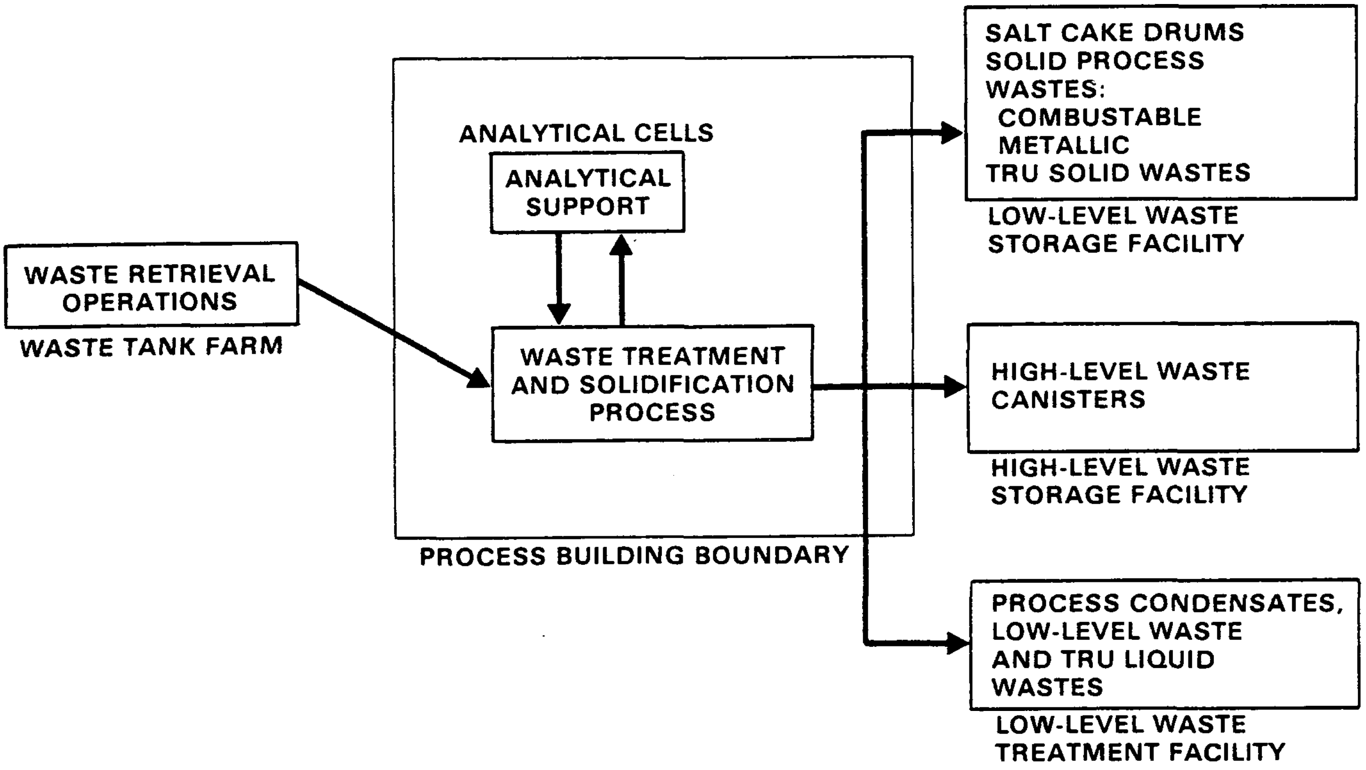

FIGURE 5.11. Process Interfaces for the Waste Treatment and Solidification Process

- Process Analytical Support Facility--This facility includes a radioactive and nonradioactive analytical laboratory that supports all process activities that occur on the plant site. Some analyses are required for compliance with environmental standards; other analyses are required for process control and process quality verification.
- Low-Level Waste Treatment Facility--This process facility treats the solidification process condensates prior to release in the environment. This facility also has packaging and solidification equipment for low-level waste and transuranic-contaminated wastes.
- Low-Level Waste Storage Facility--This facility is a storage building for the low-level waste and transuranic solid wastes generated from processing operations. Both metallic and combustible wastes are stored in this facility.

- High-level Waste Storage Facility--This facility is required for storing canisters containing high-level waste glass that are generated during the high-level waste solidification operations.

5.6 ENVIRONMENTAL CONSIDERATIONS

The waste management and waste solidification facility will be required to adhere to effluent release guidelines set by the Department of Energy, the Nuclear Regulatory Commission, the Environmental Protection Agency and the State of New York. Limits on radioactive release are controlled primarily by the Department of Energy Manual Chapter 0524 which is similar to Nuclear Regulatory Commission regulations published in 10 CFR 20 (Nuclear Regulatory Commission 1980). The Environmental Protection Agency regulations (40 CFR 190 and 40 CFR 191; Environmental Protection Agency 1980a and b) pertain only to release from the commercial nuclear fuel cycle facilities but provide guidance on acceptable limits for public exposure. Environmental Protection Agency guidelines represent future trends and should be considered when setting the solidification plant release limits.

Major environmental laws pertaining to nonradioactive releases from a nuclear waste and solidification facility are the Federal Water Pollution Control Act and the Clean Air Act. These acts have established the limits for the State of New York water quality regulations. The State of New York requires the WNYNSC to meet release standards on process streams released from the plant. The current sample frequency and release limits set by the New York State Pollution Limits Discharge Elimination System (SPDES) are summarized in Table 5.21.

TABLE 5.21. Specific WNYNSC Monitoring Samples

Sample	Frequency	Requirements
SPDES 001 discharge to Endman Brook	1/month	Trace metals, NH$_3$, suspended solids, temperature, pH, radioactivity, gross β-λ, isotopic gamma, ^{90}Sr, ^{129}I.
SPDES 002 effluent from low-level waste treatment system	1/month	Suspended solids.
SPDES 003 effluent from burial site lagoon	as required	Burial trenches before each transfer, gross β-λ, tritium, isotopic gamma, ^{90}Sr, ^{129}I.
SPDES 004 (sanitary) discharge)	1/month	Suspended solids, pH.
	1/quarter	BOD-5, suspended solids.
SPDES 005 ditch tributary of Endman Brook	1/month	Suspended solids, pH.
SPDES 006 Endman Brook downstream of SPDES 001	2/month	NH$_3$, pH, Fe.
Tank 8D-2 liquid	1/quarter	Oxalate, OH$^-$, Cs, emergency specification, miscellaneous.
Tank 8D-2 and 4 vapor	1/quarter	H$_2$ gas analysis.

6.0 PROCESS EQUIPMENT

This section identifies and describes the major processing equipment required for the reference (salt/sludge separation) process. Included is a preliminary assessment of maintenance needs, including an estimate of equipment life and spare recommendations.

6.1 EQUIPMENT DESCRIPTION

The description of the major equipment pieces, in so far as practicable, is organized to follow the eight functional systems of the process. As an aid to the reader, a listing of these systems is repeated here:

System	Function
A	Centrifugation
B	Agglomeration, settling and filtration
C	Cesium ion exchange and concentration
D	Strontium ion exchange
E	Calcination, vitrification and canister handling
F	Recycle and secondary evaporation
G	Calciner effluent treatment (off gas)
H	Salt solidification

Supporting pieces of equipment, such as tanks, pumps and condensers, that are applicable to more than one system are described at the end of this subsection.

6.1.1 Centrifugation (System A)

The first step in clarifying the alkaline waste is to process the slurry through two successive centrifugations (CE-A-1 & CE-A-2).[a] The design chosen for the centrifuges was recommended by the Savannah River Laboratory

Design data for the centrifuges are shown in Figure 6.1. The descriptive design elements of the centrifuge configuration are: 1) basket centrifuge,

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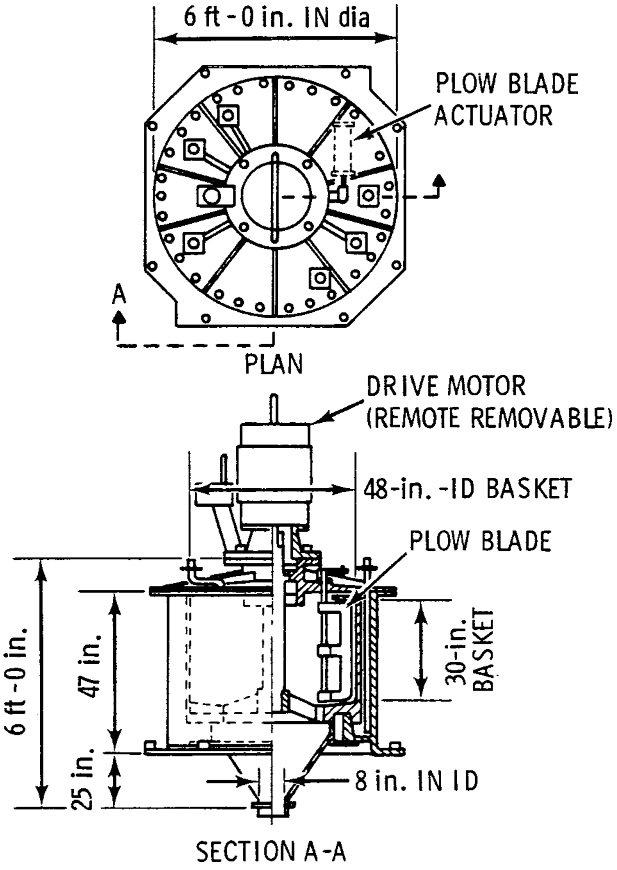
[a] Designations in parentheses here and elsewhere in this section identify equipment pieces as shown on equipment data sheets in Appendix B.



DESIGN DATA:

CODE: MANUFACTURERS STANDARD
MATERIAL: TYPE 304-L STAINLESS STEEL
DESIGN TEMPERATURE: 35°C (95°F)
DESIGN PRESS: ATMOSPHERIC
OPERATING PRESS: -1 in. H_2O (VENT)
FLUID pH: 7 TO 8
FLUID DENSITY: 1.55 g/cm^3
BASKET DIAMETER: 48 in.
BASKET DEPTH: 30 in.
BASKET RPM: 1400
ESTIMATED WEIGHT EMPTY: 10,000 lb
ESTIMATED WEIGHT AT OPERATING CAPACITY: 12,000 lb

FIGURE 6.1. Centrifuge

2) automated clarification, 3) top drive, two bearing, 4) vertical shaft, 5) cantilevered shaft, 6) bottom discharge, and 7) direct coupled.

Totally enclosed, fan-cooled 1200-rpm electric motors are used to power the centrifuges. The motors are mounted directly on the cover; the cover, in turn, is attached to the centrifuge case by means of special isolaters. The motor can be replaced by remote means.

The bottom of the centrifuge bowl slopes toward the center to assure free drainage. To avoid small pockets where contamination can accumulate, the inside corners of the bowl are filleted and all surfaces are smooth. In general, the bowl is designed in accordance with 3-A and GMP guidelines (dairy and pharmaceutical, respectively). The discharge opening in the bottom of the bowl is bridged by radial arms that are incorporated into a stainless steel casting called the "spider."

The cake is removed from the centrifuge bowl with a hydraulically actuated plow. The cake falls out the bottom of the bowl and through an 8-in.-dia pipe into a slurry catch tank (A-3 or A-5) that is located directly below the centrifuge.

6.1.2 Agglomeration, Settling and Filtration (System B)

The centrate requires additional clarification before being fed to the ion-exchange columns. This is accomplished by agglomeration and settling, followed by filtration.

Gravity Settlers

There are two gravity settlers (GS-B-1 and GS-B-2), which operate in parallel to attain the required processing rate. Design data for the gravity settlers are shown in Figure 6.2. They are standard design, stainless steel tanks, equipped with agitators and temperature controls. Following completion of the agglomeration/settling cycles, steam jets are used to decant the supernate and to transfer the solids.

Sand Filters

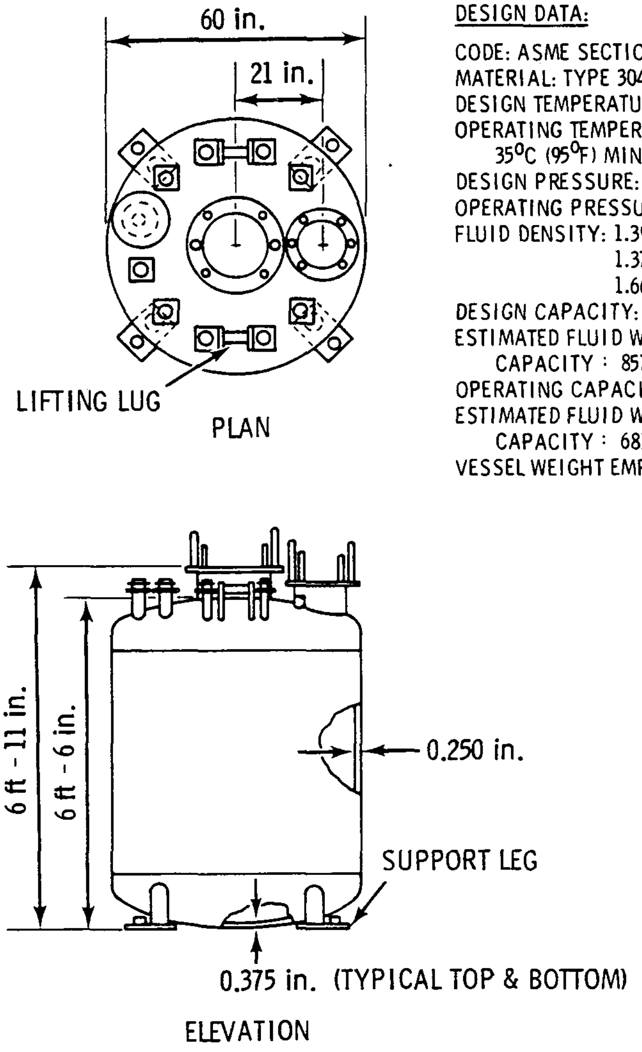
The supernate from the gravity settlers is passed through two successive sand filters (F-B-1 and F-B-2). Design data for these filters are shown in Figure 6.3. Liquid flows through the filters by gravity. A steam jet is used to remove the loaded filter media.

6.1.3 Ion Exchange (Systems C and D)

The supernate from the second sand filter is pumped through four successive ion-exchange columns to remove the cesium, strontium and actinides.

Ion-Exchange Columns

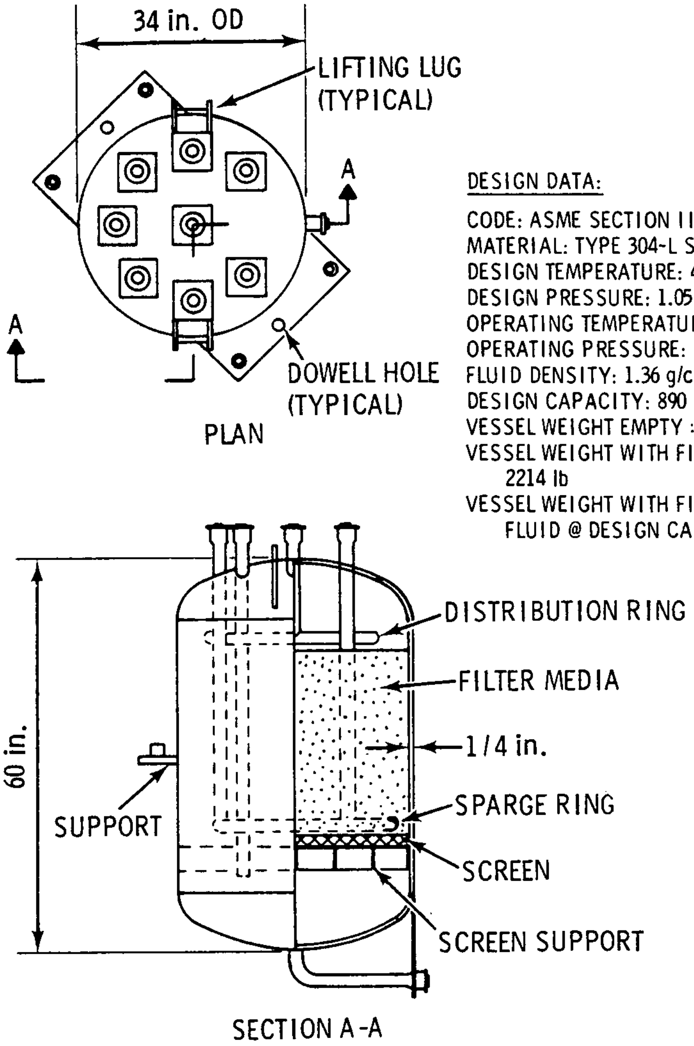
A typical ion-exchange column is shown in Figure 6.4. The columns are standard, Schedule-40, 304L stainless steel pipe and are designed for complete remote replacement. Of necessity, these ion-exchange columns are operated at a positive pressure; consequently, special considerations will be required in subsequent design efforts to assure that no path exists for forcing contamination from the columns into the operating aisles.



DESIGN DATA:

CODE: ASME SECTION III CLASS 2
MATERIAL: TYPE 304-L STAINLESS STEEL
DESIGN TEMPERATURE: 105°C (221°F)
OPERATING TEMPERATURE: 95°C MAX,
35°C (95°F) MIN
DESIGN PRESSURE: 1.05 kg/cm^2 (15 psi)
OPERATING PRESSURE: -3 in. H_2O
FLUID DENSITY: 1.39 g/cm^3 INLET
1.37 g/cm^3 DECANT
1.66 g/cm^3 SOLIDS
DESIGN CAPACITY: 2800 L (740 gal)
ESTIMATED FLUID WEIGHT @ DESIGN
CAPACITY : 8570 lb
OPERATING CAPACITY : 2225 L (588 gal)
ESTIMATED FLUID WEIGHT @ OPERATING
CAPACITY : 6810 lb
VESSEL WEIGHT EMPTY : 2300 lb

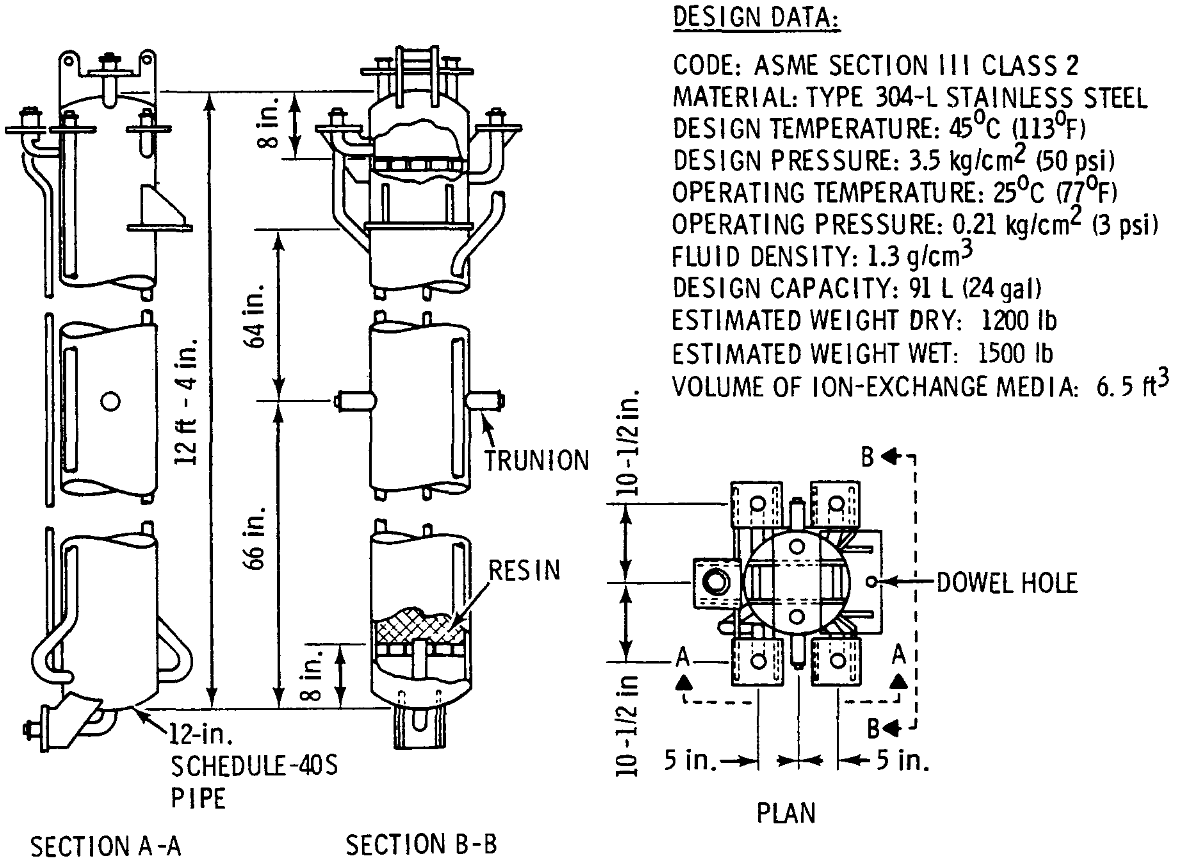
FIGURE 6.2. Gravity Settlers



DESIGN DATA:

CODE: ASME SECTION III CLASS 2 VESSEL
MATERIAL: TYPE 304-L STAINLESS STEEL
DESIGN TEMPERATURE: 45°C (77°F)
DESIGN PRESSURE: 1.05 kg/cm^2 (15 psi)
OPERATING TEMPERATURE: 25°C (77°F)
OPERATING PRESSURE: -3 in. H_2O
FLUID DENSITY: 1.36 g/cm^2
DESIGN CAPACITY: 890 L (235 gal)
VESSEL WEIGHT EMPTY : 614 lb
VESSEL WEIGHT WITH FILTER MEDIA : 2214 lb
VESSEL WEIGHT WITH FILTER MEDIA AND FLUID @ DESIGN CAPACITY : 3340 lb

FIGURE 6.3. Sand Filters


DESIGN DATA:

CODE: ASME SECTION III CLASS 2
MATERIAL: TYPE 304-L STAINLESS STEEL
DESIGN TEMPERATURE: 45°C (113°F)
DESIGN PRESSURE: 3.5 kg/cm^2 (50 psi)
OPERATING TEMPERATURE: 25°C (77°F)
OPERATING PRESSURE: 0.21 kg/cm^2 (3 psi)
FLUID DENSITY: 1.3 g/cm^3
DESIGN CAPACITY: 91 L (24 gal)
ESTIMATED WEIGHT DRY: 1200 lb
ESTIMATED WEIGHT WET: 1500 lb
VOLUME OF ION-EXCHANGE MEDIA: 6.5 ft^3

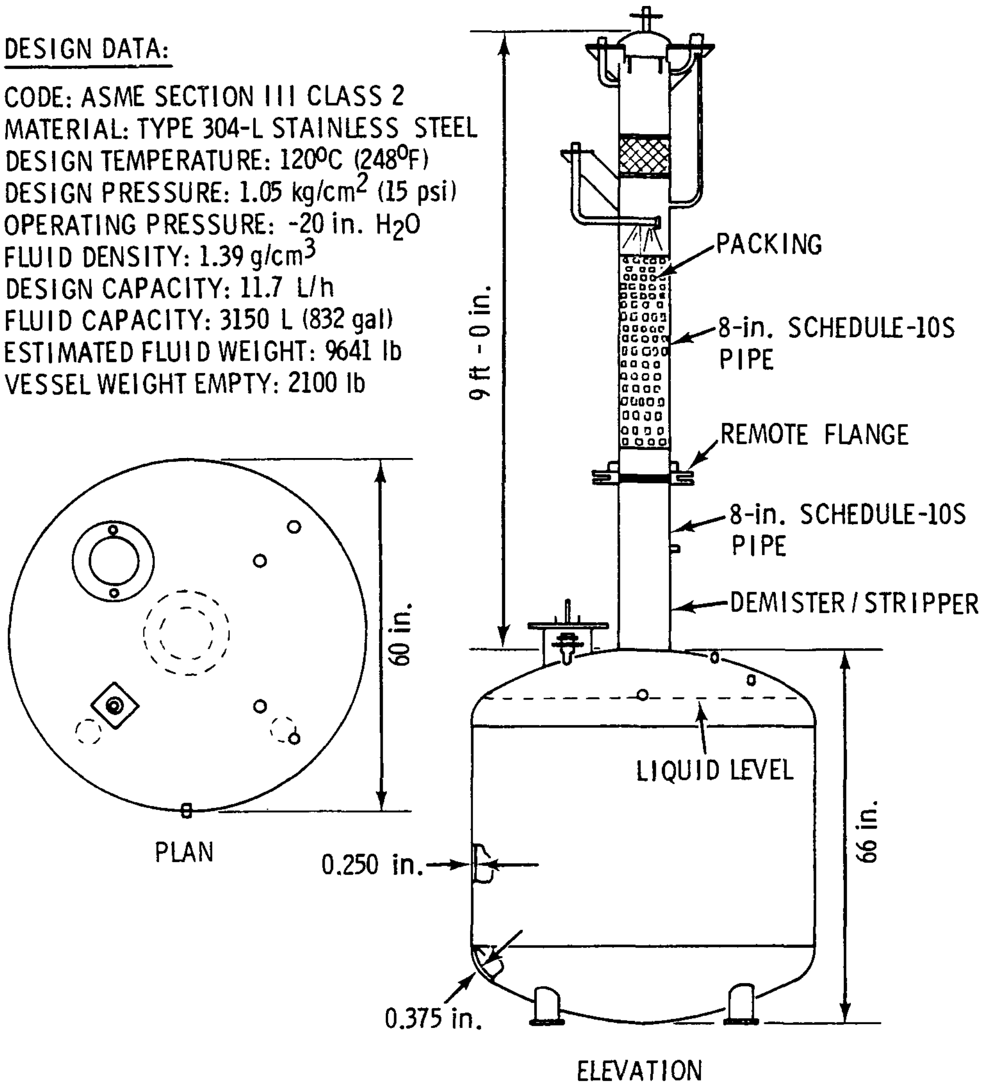
FIGURE 6.4. Typical Ion-Exchange Column

Concentrator for Cesium Eluate

A reboiler (pot) type concentrator (CR-C-1) is used to concentrate the eluate from the cesium ion-exchange columns. Design data for this unit are shown in Figure 6.5. A packed tower and a stripper/demister are used for reclaiming ammonia. Heat to the reboiler is provided by a steam-heated tube bundle that is replaceable by remote means.

6.1.4 Calcination, Vitrification and Canister Handling (System E)

This section includes those equipment pieces that accomplish the main purpose of the solidification program, i.e., incorporating the radioactive components of the high-level waste into a borosilicate glass.



DESIGN DATA:

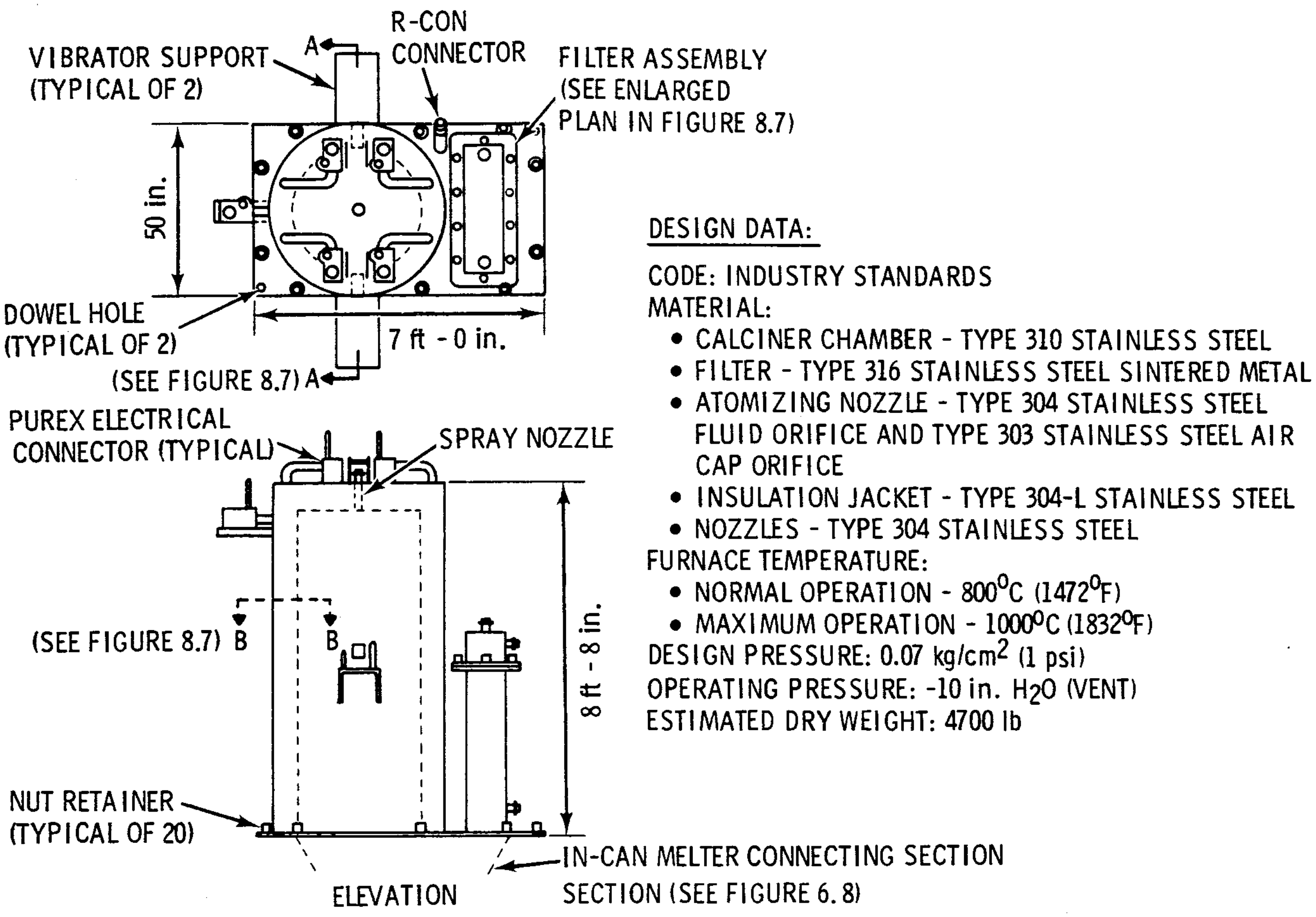
CODE: ASME SECTION III CLASS 2
MATERIAL: TYPE 304-L STAINLESS STEEL
DESIGN TEMPERATURE: 120°C (248°F)
DESIGN PRESSURE: 1.05 kg/cm^2 (15 psi)
OPERATING PRESSURE: -20 in. H_2O
FLUID DENSITY: 1.39 g/cm^3
DESIGN CAPACITY: 11.7 L/h
FLUID CAPACITY: 3150 L (832 gal)
ESTIMATED FLUID WEIGHT: 9641 lb
VESSEL WEIGHT EMPTY: 2100 lb

FIGURE 6.5. Concentrator

Spray Calciner

Design data for the spray calciner are shown in Figures 6.6 and 6.7. The design is based on extensive development activities that have been performed by Pacific Northwest Laboratory to which standard Purex remote technology has been applied.

The spray (feed) nozzle is a modified, commercial atomizing nozzle incorporated into a 2-in. Purex pipe connector. Other replaceable components include the: 1) resistance-heated furnace, 2) extended-shaft vibrators (two), and 3) a 65-μm filter assembly.



DESIGN DATA:

CODE: INDUSTRY STANDARDS

MATERIAL:

- CALCINER CHAMBER - TYPE 310 STAINLESS STEEL
- FILTER - TYPE 316 STAINLESS STEEL SINTERED METAL
- ATOMIZING NOZZLE - TYPE 304 STAINLESS STEEL FLUID ORIFICE AND TYPE 303 STAINLESS STEEL AIR CAP ORIFICE
- INSULATION JACKET - TYPE 304-L STAINLESS STEEL
- NOZZLES - TYPE 304 STAINLESS STEEL

FURNACE TEMPERATURE:

- NORMAL OPERATION - 800°C (1472°F)
- MAXIMUM OPERATION - 1000°C (1832°F)

DESIGN PRESSURE: 0.07 kg/cm^2 (1 psi)

OPERATING PRESSURE: -10 in. H_2O (VENT)

ESTIMATED DRY WEIGHT: 4700 lb

FIGURE 6.6. Spray Calciner

Frit Feeder

Glass formers (frit) are added to the cone section of the spray calciner where they mix with the calcine. A commercial, weight-belt feeder is used to meter the frit. A special air-lock valve assembly in the frit-addition line, located in a shielded wall niche (see drawing SK-7-967 in Appendix A), prevents back flow of contamination from the spray calciner into the operating isles.

Spray Calciner/In-Can Melter Connection Assembly

A multipurpose assembly is required for connecting the spray calciner to the two in-can melters. Design data for this connection assembly are shown in Figure 6.8.

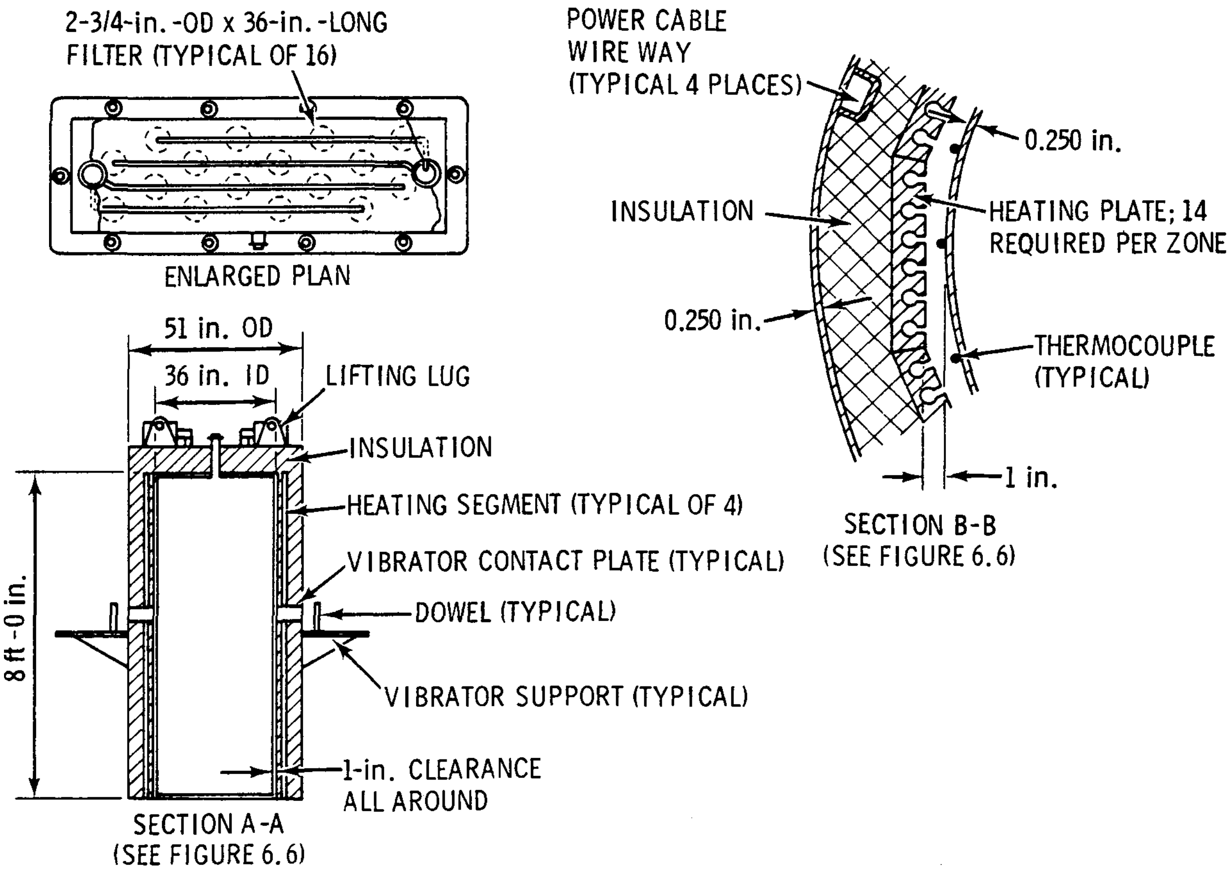


FIGURE 6.7. Spray Calciner Details

At the top of the assembly is a special valve for diverting the calcine/frit mixture to the appropriate in-can melter. The valve is designed to provide the necessary ventilation simultaneously to both melters, irrespective of the direction of calcine/frit flow.

Rotating joints are incorporated into each of the two chutes for feeding calcine to the canister. The lower end of each chute has a retractable spout. At the end of this spout is a retractable cone. When a canister is filled, the cone is retracted (with a jack screw) against the spout to effect a closure at the lower end of the feed chute. As the cone is further retracted, the spout is raised out of the canister allowing the feed calcine chute to.be rotated so that the filled canister can be removed from the in-can melter furnace.

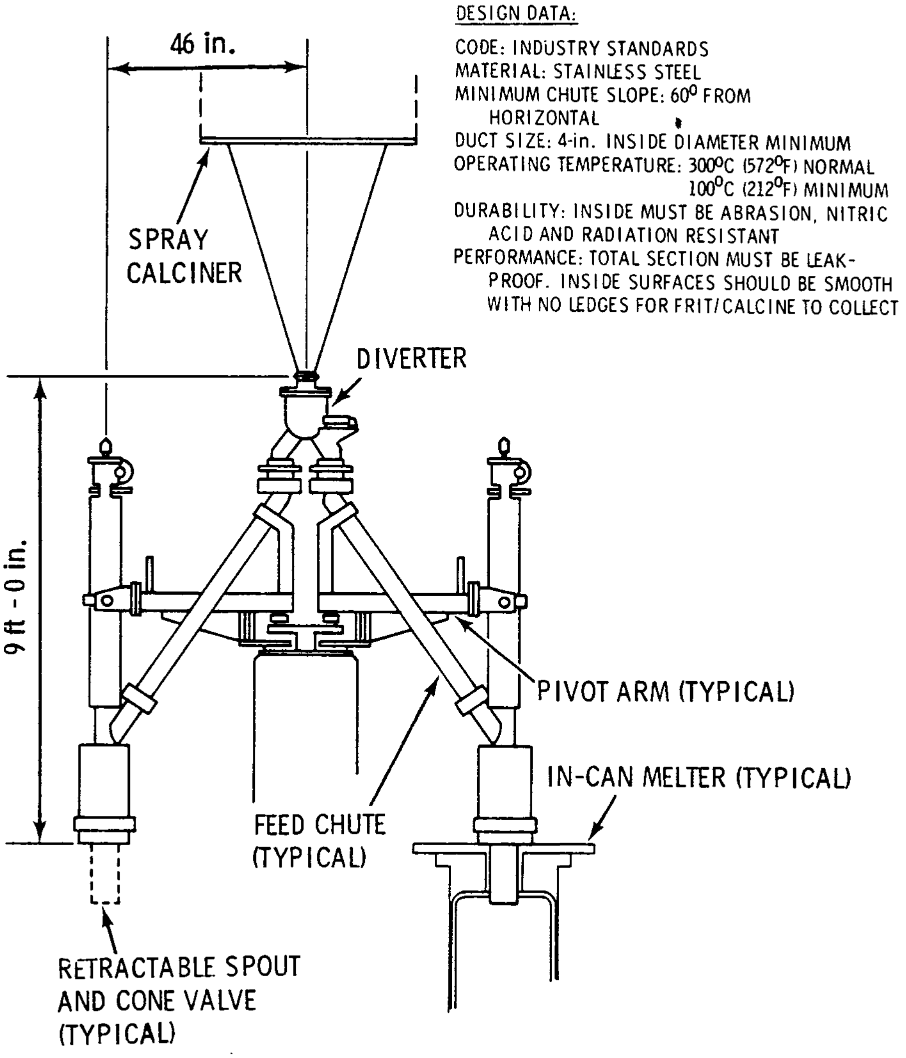


FIGURE 6.8. Spray Calciner/In-Can Melter Connection Section

A secondary function of the retractable cone is to distribute the calcine/frit mixture evenly in the canister to avoid a buildup of calcine/frit mixture on top of the fin assembly that is in the canister (see Figure 6.11).

In-Can Melters

There are two in-can melters (in-can melter E-1 and in-can melter E-2). Design data for these furnaces are shown in Figures 6.9 and 6.10. The canisters are loaded into and removed from the furnaces at the top. The canisters are heated by 200-kW, resistance-heated furnaces. Each furnace is divided into

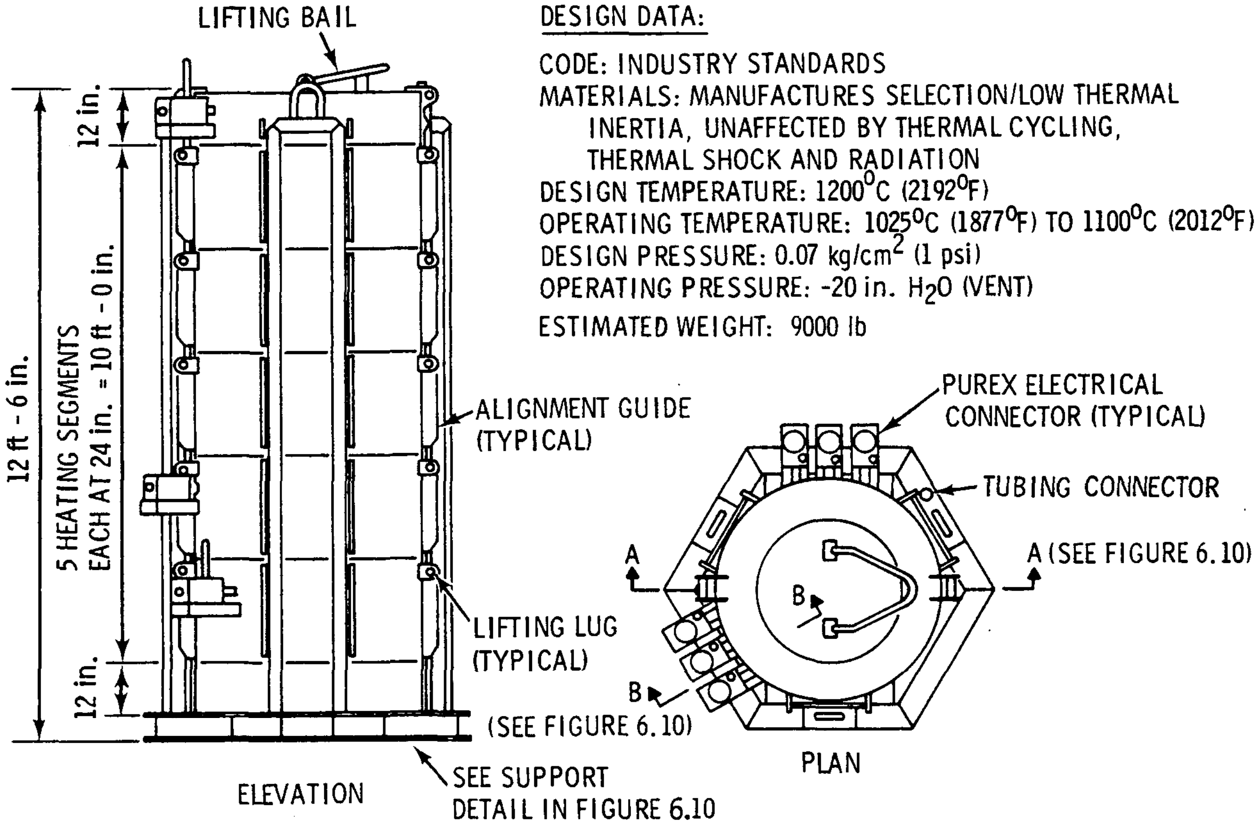


FIGURE 6.9. In-Can Melter

five 40-kW, 24-in.-long heating segments (zones). An argon purge into the furnace basket helps prevent excessive spalling of the processing canister.

The furnace segments are placed in a support (see Figure 6.9). The support containing the segments is mounted on load cells that provide one method of determining fill level in the canister. Other methods used to determine fill level include: 1) gamma scan, 2) temperature and 3) power consumption.

Canister

The high-level waste canister is shown in Figure 6.11. It is constructed from 24-in. Schedule-40, 304L stainless steel pipe. The bottom end is closed with a standard flanged and dished head. The upper end features a "twist lock" closure developed by Pacific Northwest Laboratory. An appropriate lifting pintle is included for remote handling of the canister. During processing, the canister is supported from the bottom.

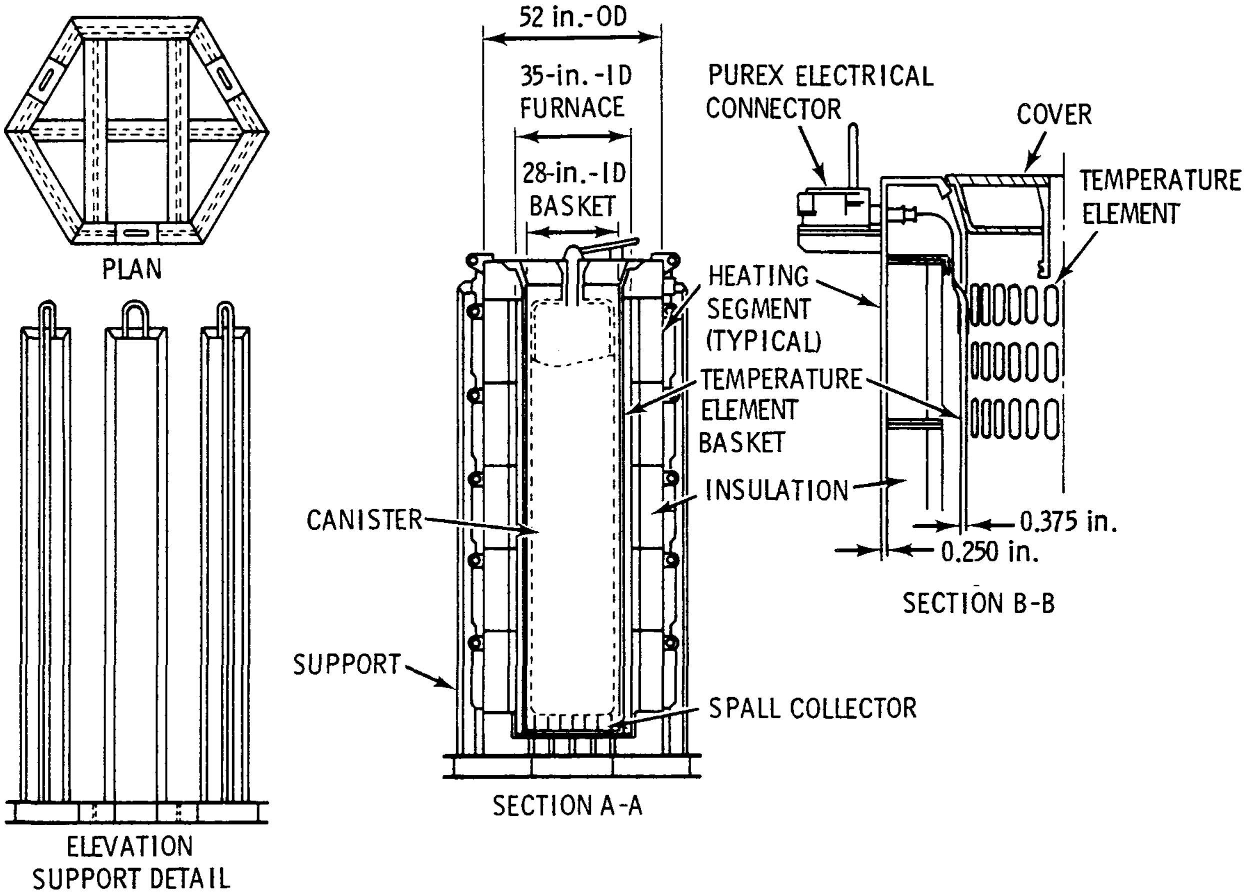


FIGURE 6.10. In-Can Melter Details

A drop-in fin assembly is incorporated into the canister design to evenly distribute the heat during processing and subsequent storage.

Lid-Welding and Leak-Testing Station

After processing has been completed in the in-can melter, the high-level waste canisters are moved to the weld/leak-test station where a lid is welded onto the canister. A helium leak test is then performed on the lid weld. Design data for the weld/leak-test station are shown in Figure 6.12.

The lid welder design is based on development work that has been performed at Pacific Northwest Laboratory. A similar welder has been fabricated and extensively tested on twist-lock lids like those specified for the reference design canisters. With this design, the canister remains stationary and only the weld torch assembly is rotated. The capability for repairing welds is included in the station design.

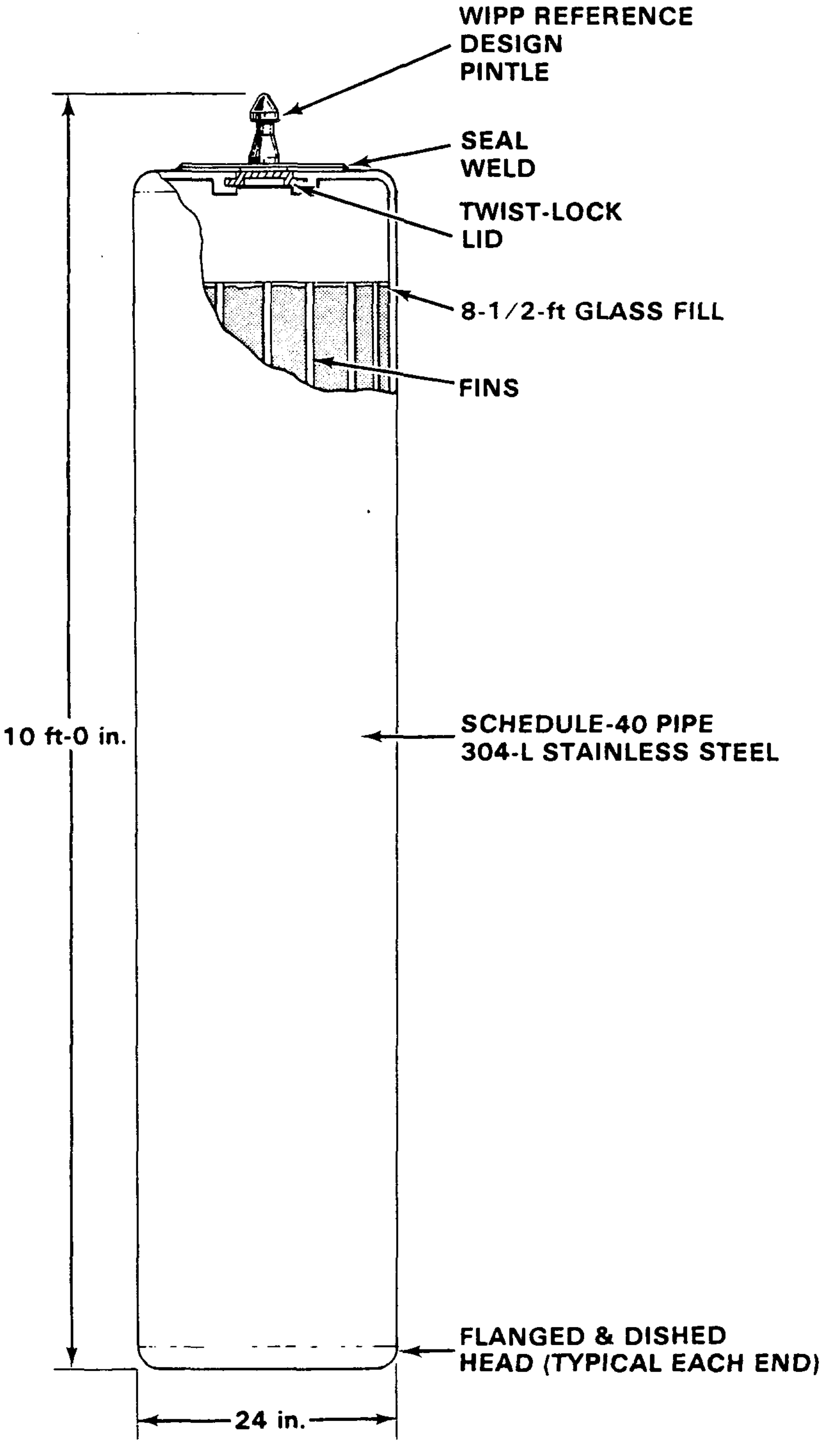


FIGURE 6.11. High-Level Waste Canister

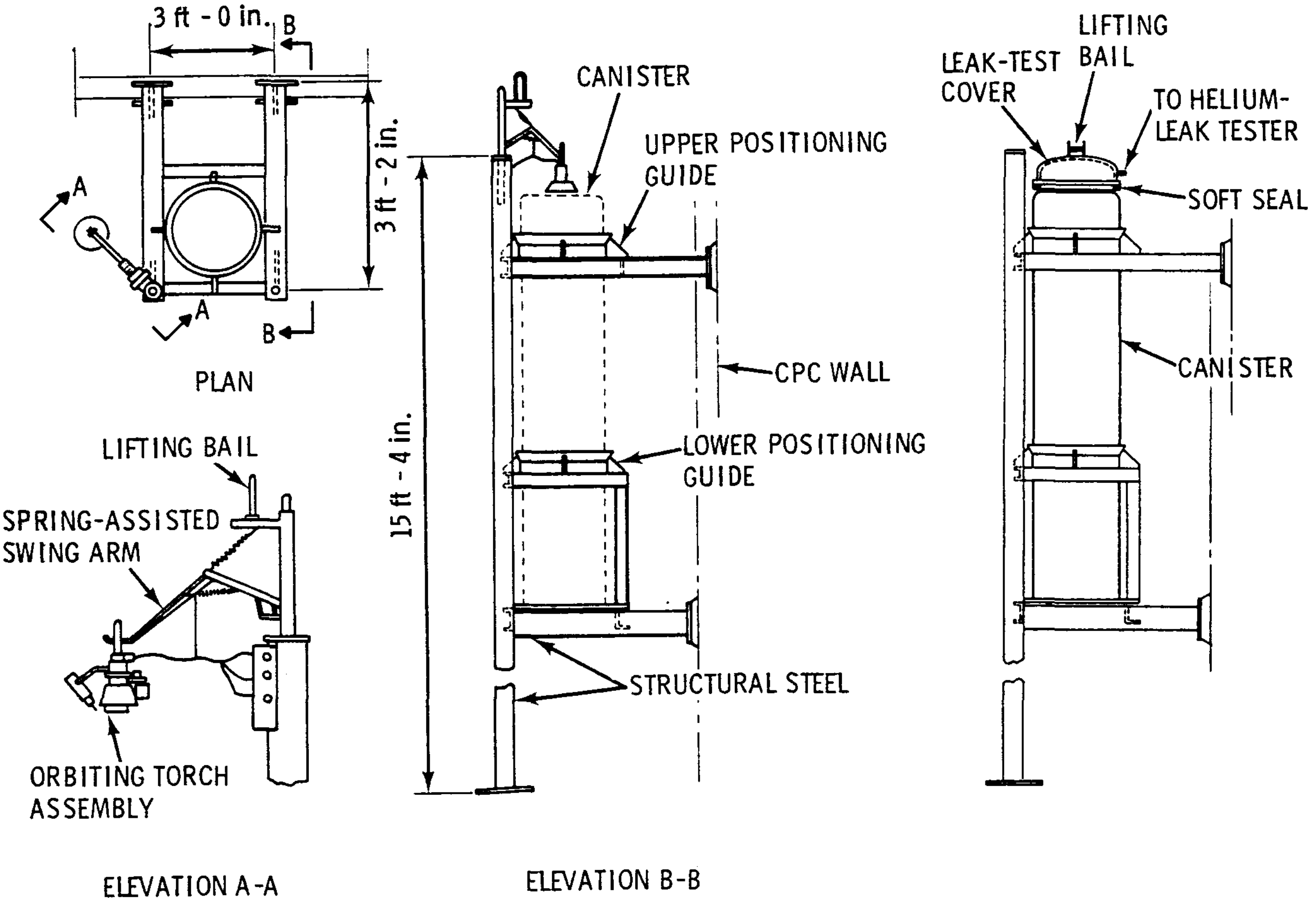
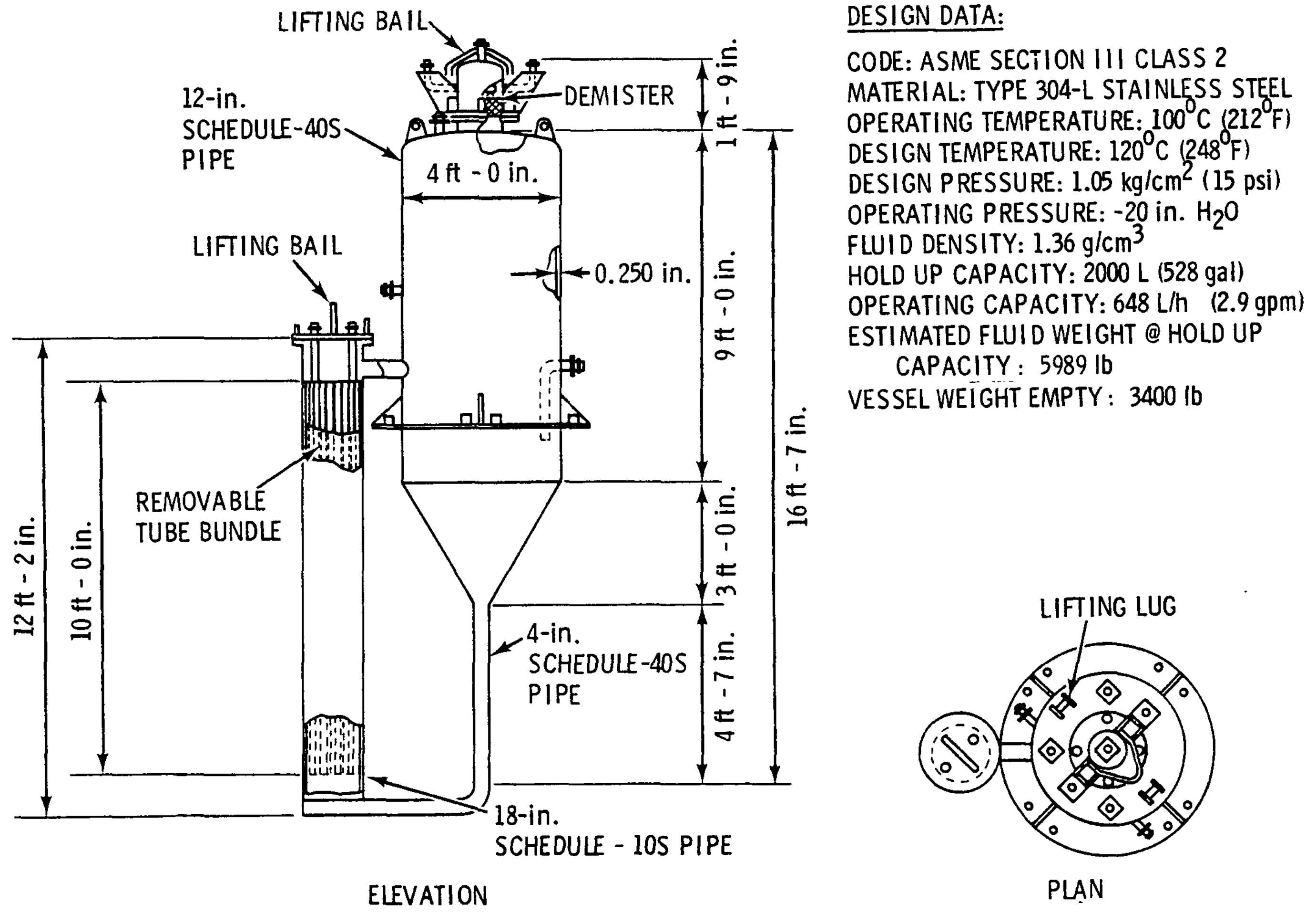


FIGURE 6.12. High-level Waste Canister Weld/Leak-Test Station

A slow-release helium source is placed in the canister before installing the canister lid. When the lid weld is completed, a leak-test cover (bell jar) is placed over the weld area. A soft (easily replaceable) gasket is placed between the cover and the top of the canister. The weight of the cover and the subsequent vacuum that is applied provides an effective seal. The leak test is done by evacuating the inside of the cover and analyzing the gas stream with a helium mass spectrometer.

6.1.5 Recycle and Secondary Evaporation (System F)

Thermo-syphon evaporators (E-F-1 and E-F-2) are used for concentrating the various recycle streams. Design data for a typical evaporator are shown in Figure 6.13. These evaporators are steam-heated. The tube bundles in E-F-1 and E-F-2 are designed for remote replacement.



DESIGN DATA:

CODE: ASME SECTION III CLASS 2
MATERIAL: TYPE 304-L STAINLESS STEEL
OPERATING TEMPERATURE: 100°C (212°F)
DESIGN TEMPERATURE: 120°C (248°F)
DESIGN PRESSURE: 1.05 kg/cm^2 (15 psi)
OPERATING PRESSURE: -20 in. H_2O
FLUID DENSITY: 1.36 g/cm^3
HOLD UP CAPACITY: 2000 L (528 gal)
OPERATING CAPACITY: 648 L/h (2.9 gpm)
ESTIMATED FLUID WEIGHT @ HOLD UP CAPACITY: 5989 lb
VESSEL WEIGHT EMPTY: 3400 lb

FIGURE 6.13. Typical Thermo-Syphon Evaporator

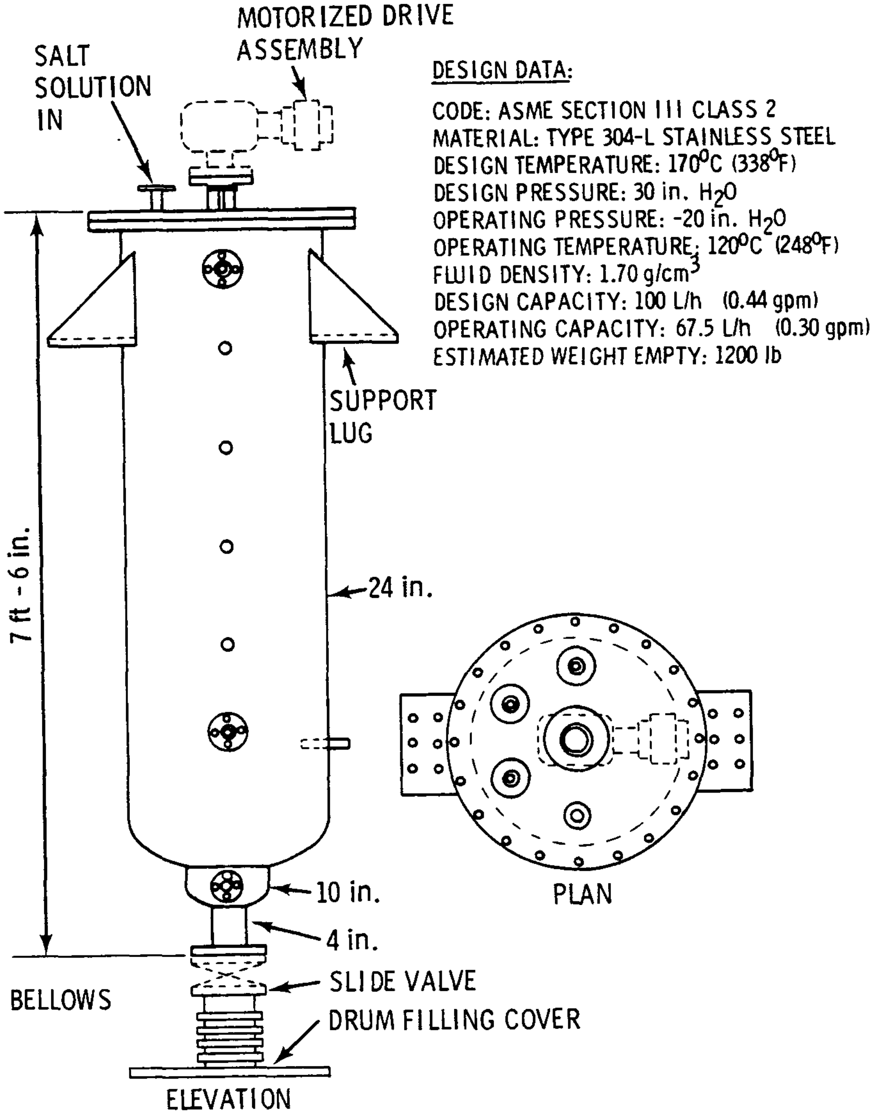
6.1.6 Calciner Effluent Treatment (System G)

The off gas from the spray calciner passes through a venturi scrubber/cyclone unit (J-G-1), a downdraft condenser (C-G-1), a demister (D-G-1), a silver mordenite adsorbent bed (AB-G-2), a silica gel adsorbent bed (AB-G-1), a deep-bed gas filter (F-G-1), high efficiency particulate air filters, and a NO_x destructor (CB-G-1). It is then released to the stack.

6.1.7 Salt Solidification (System H)

A thermo-syphon evaporator (E-H-1) is used for the first step in concentrating the low-level salt solution. Figure 6.13 shows a typical evaporator.

A vertical wiped-film evaporator (E-H-2) is specified for final dewatering of the salt solution. Design data for this evaporator are shown in Figure 6.14. The design is based on a commercial unit; it has been modified as required for its intended use. This evaporator will be contact-installed and contact-maintained; consequently, no remote features are included.



DESIGN DATA:

CODE: ASME SECTION III CLASS 2
MATERIAL: TYPE 304-L STAINLESS STEEL
DESIGN TEMPERATURE: 170°C (338°F)
DESIGN PRESSURE: 30 in. H_2O
OPERATING PRESSURE: -20 in. H_2O
OPERATING TEMPERATURE: 120°C (248°F)
FLUID DENSITY: 1.70 g/cm^3
DESIGN CAPACITY: 100 L/h (0.44 gpm)
OPERATING CAPACITY: 67.5 L/h (0.30 gpm)
ESTIMATED WEIGHT EMPTY: 1200 lb

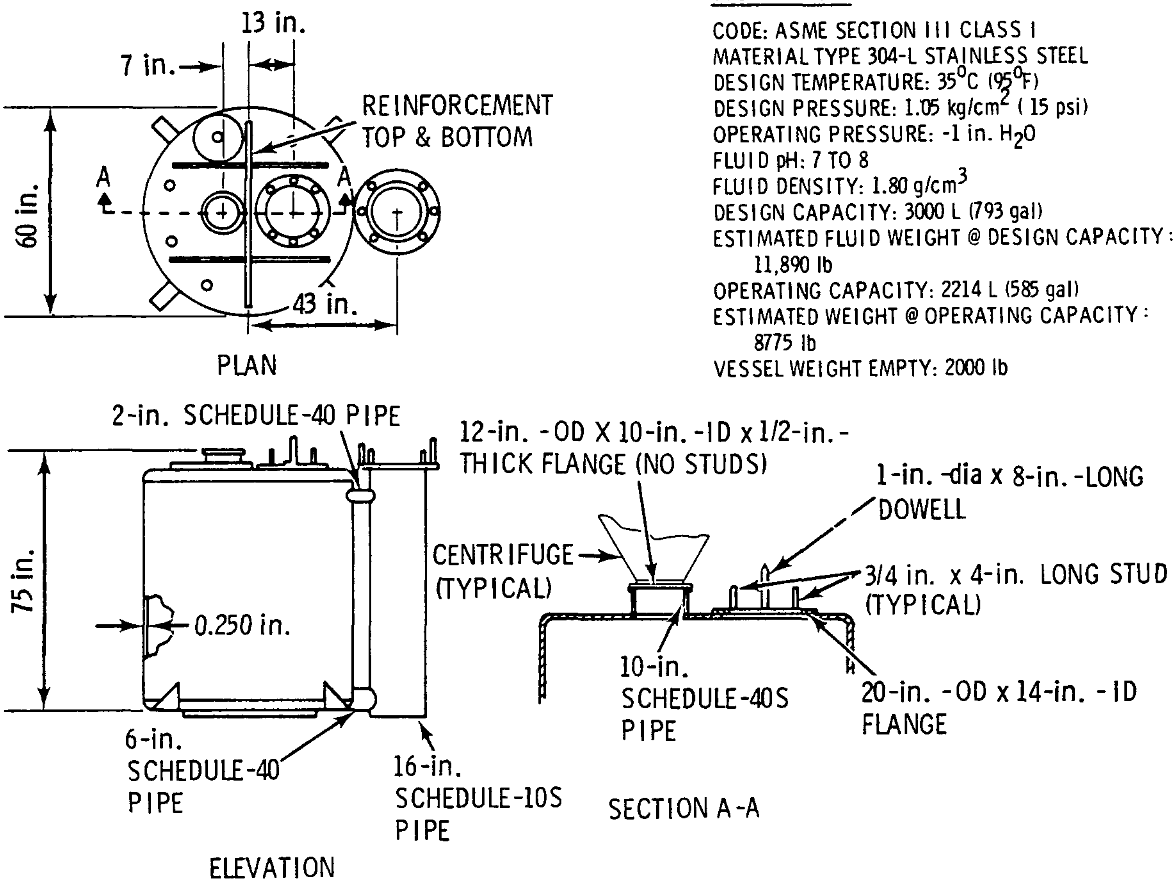
FIGURE 6.14. Wiped-Film Evaporator

6.1.8 Supporting Process Equipment

The equipment discussed here supports all the functional systems.

Process Tanks

The design of the process tanks that are to be installed in the Chemical Processing Cell is based on standard Purex remote technology . Design data for a typical process tank are shown in Figure 6.15. The side chamber shown is for installing either a cantilevered or submerged pump. This chamber is included only where required for overhead access or to provide room for both an agitator and a top-entry pump. In addition to standard instrumentation, the tanks are serviced by air spargers, vortex preventers (baffles), and cooling/heating coils as dictated by the processing requirements.



DESIGN DATA:

CODE: ASME SECTION III CLASS I
MATERIAL TYPE 304-L STAINLESS STEEL
DESIGN TEMPERATURE: 35°C (95°F)
DESIGN PRESSURE: 1.05 kg/cm^2 (15 psi)
OPERATING PRESSURE: -1 in. H_2O
FLUID pH: 7 TO 8
FLUID DENSITY: 1.80 g/cm^3
DESIGN CAPACITY: 3000 L (793 gal)
ESTIMATED FLUID WEIGHT @ DESIGN CAPACITY: 11,890 lb
OPERATING CAPACITY: 2214 L (585 gal)
ESTIMATED WEIGHT @ OPERATING CAPACITY: 8775 lb
VESSEL WEIGHT EMPTY: 2000 lb

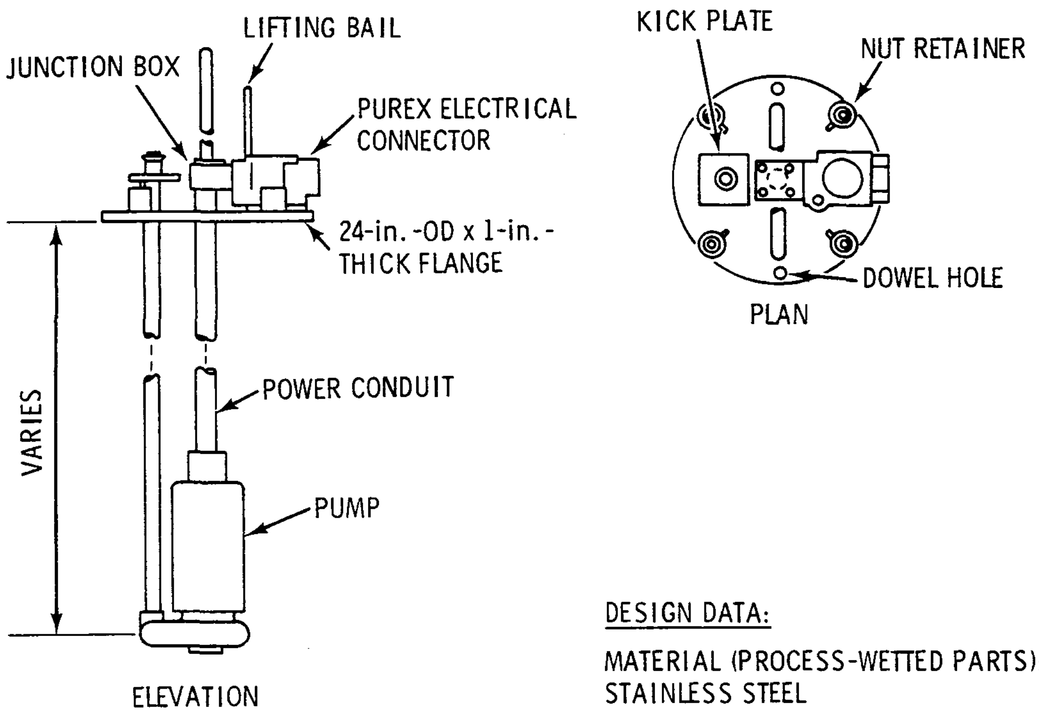
FIGURE 6.15. Typical Process Tank

Process tanks for use in the final salt concentration system (in the Scrap Removal Room) are similar in design to the Chemical Processing Cell process tanks except they are designed for direct handling during installation and removal. Remote features are limited to those necessary for remote operation and routine maintenance.

Process Pumps

Two basic types of electric-driven centrifugal pumps are used. The pumps are mounted on flanges designed for remote replacement. Variable-speed motors are used on some of the pumps to control flow rate. Figure 6.16 shows a typical submerged pump (canned) that is specified for pumping clear solutions. With this pump, the process solution, after passing through a hydroclone, is used to lubricate the pump bearings.

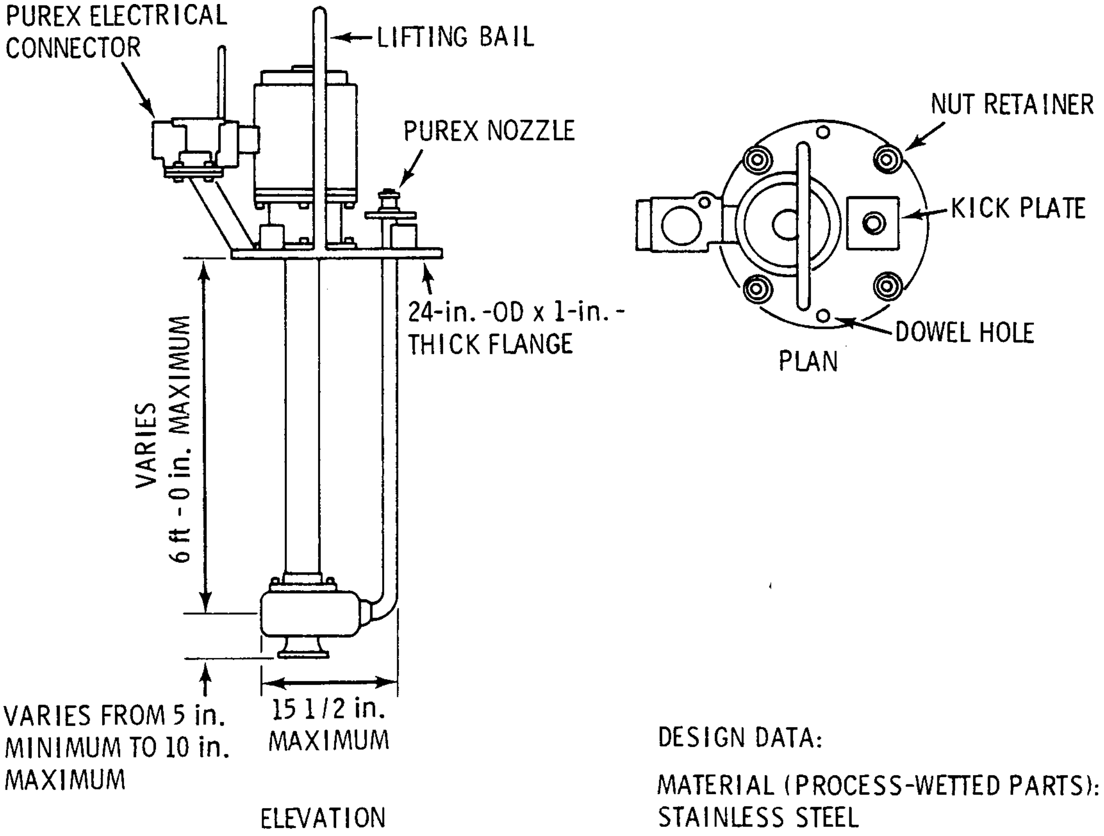
Figure 6.17 shows a typical cantilever pump. This type of pump is also submerged in the process solution during operation. However, as it has no bearings located below the solution level, it is specified for pumping slurries and other solutions containing abrasive solids.



DESIGN DATA:

MATERIAL (PROCESS-WETTED PARTS): STAINLESS STEEL

FIGURE 6.16. Typical Submerged Pump Assembly



DESIGN DATA:

MATERIAL (PROCESS-WETTED PARTS):
STAINLESS STEEL

FIGURE 6.17. Typical Cantilever Pump Assembly

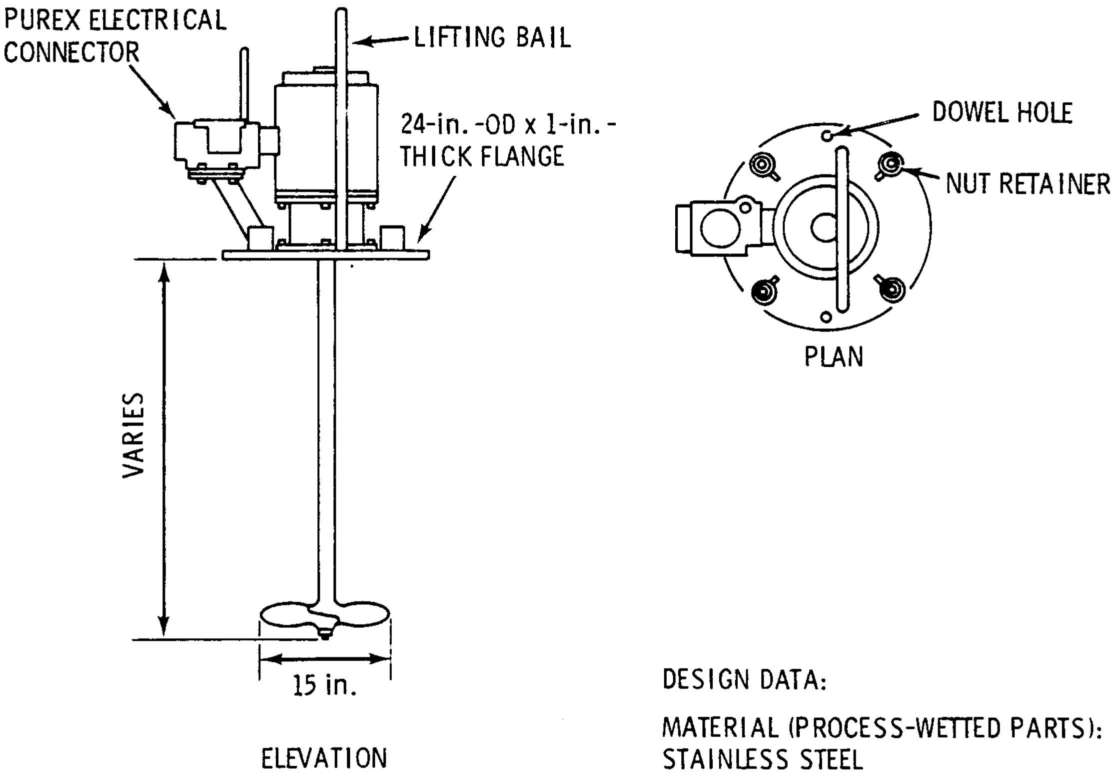
In addition to the two basic pumps described above, several conventional centrifugal pumps and one gear pump are used for pumping the various process solutions in the Scrap Removal Room.

Agitators

A typical agitator is shown in Figure 6.18. The agitators are mounted on flanges that are designed for remote replacement. They are either fixed-speed or variable-speed electric motors, as required to meet process requirements.

Condensers

Both vertical and horizontal condensers are specified. Typical design data are shown in Figures 6.19 and 6.20. All of the condensers are the tube-and-shell type. Bellows are provided in the shell for expansion. Pall rings are placed in the tubes of the vertical condensers to increase their capacity.



DESIGN DATA:

MATERIAL (PROCESS-WETTED PARTS): STAINLESS STEEL

FIGURE 6.18. Typical Agitator Assembly

Process Sampling

A new shielded room, complete with master-slave manipulators, is provided for sampling the process solutions contained in the Chemical Processing Cell tanks. Details of this room (sample blister) are shown on drawing SK-7-960 (Appendix A). A new transfer mechanism is also provided to transfer the samples from the new sample room to the existing sample conveyor that is used for moving the samples to the analytical laboratory.

Process sampling in the Scrap Removal Room is performed in-cell with the master-slave manipulators. The samples are placed in any required secondary containment and shielding and are then transported manually to the analytical laboratory.

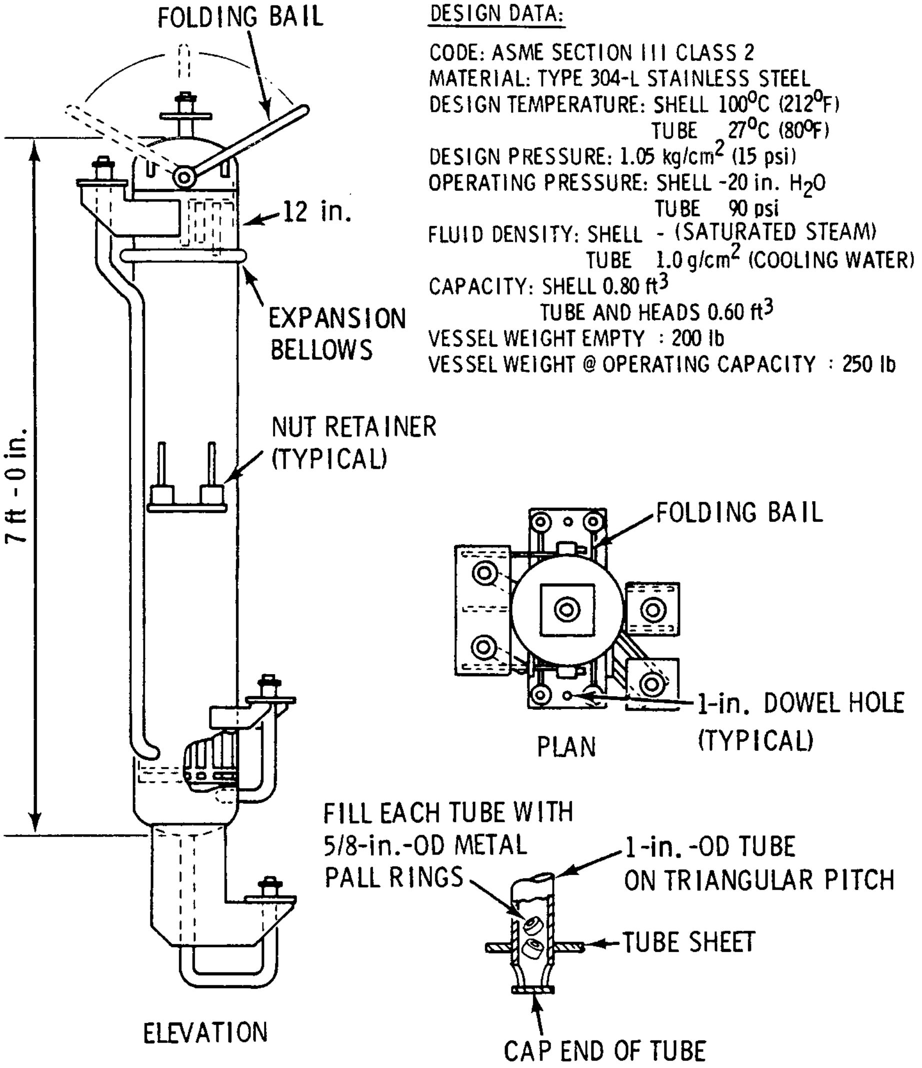
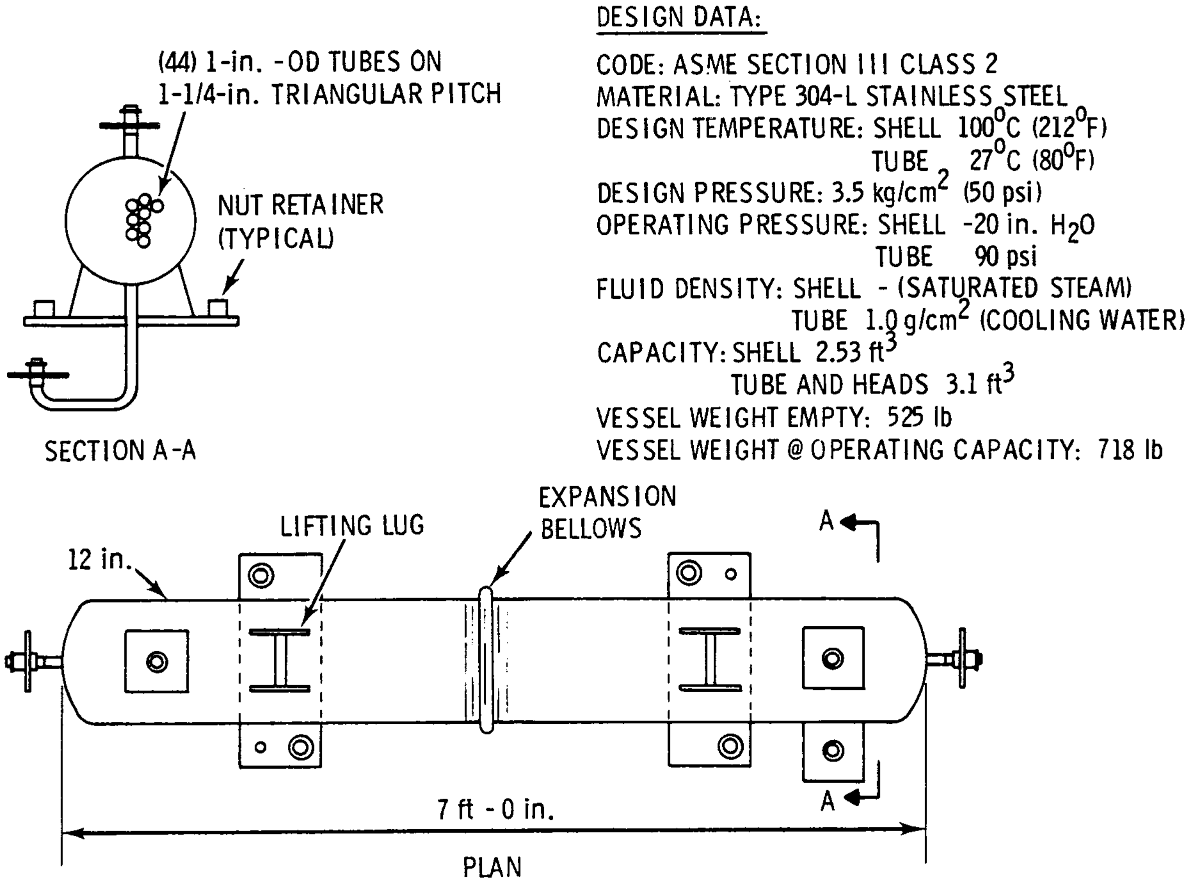


FIGURE 6.19. Typical Vertical Condenser



DESIGN DATA:

CODE: ASME SECTION III CLASS 2
MATERIAL: TYPE 304-L STAINLESS STEEL
DESIGN TEMPERATURE: SHELL 100°C (212°F)
TUBE 27°C (80°F)
DESIGN PRESSURE: 3.5 kg/cm^2 (50 psi)
OPERATING PRESSURE: SHELL -20 in. H_2O
TUBE 90 psi
FLUID DENSITY: SHELL - (SATURATED STEAM)
TUBE 1.0 g/cm^2 (COOLING WATER)
CAPACITY: SHELL 2.53 ft^3
TUBE AND HEADS 3.1 ft^3
VESSEL WEIGHT EMPTY: 525 lb
VESSEL WEIGHT @ OPERATING CAPACITY: 718 lb

FIGURE 6.20. Typical Horizontal Condenser

6.2 MAINTENANCE

The processing equipment located in the Chemical Processing Cell is designed for complete remote operation and maintenance. A remotely operated crane, one electro-mechanical manipulator and three sets of master-slave manipulators are provided for performing the remote maintenance. Oil-filled, lead-glass viewing windows and through-wall periscopes provide adequate in-cell viewing. Remote control instrumentation is employed where practicable. A few operations such as canister handling, sampling, etc., are performed manually with the mechanical equipment that is provided to assist remote operations.

The processing equipment located in the Scrap Removal Room is designed for remote operation and contact maintenance; however, some minor remote maintenance can be performed with the master-slave manipulators and the remotely operated crane. In-cell viewing is provided by two oil-filled, lead-glass windows. Where practicable, remote control instrumentation is also employed on the Scrap Removal Room equipment.

Maintenance of the Chemical Processing Cell equipment will be performed on both a scheduled and an as-needed nonscheduled basis. A plant operating efficiency of 60% is assumed. This provides the equivalent of 21 weeks/year for scheduled and nonscheduled maintenance. In general, it is desirable to operate the entire process in long-duration campaigns with scheduled shutdowns for maintenance.

Standard Hanford remote technology is incorporated into the equipment design. Processing equipment in the Chemical Processing Cell can be replaced remotely by the use of the overhead cranes. Positioning trunnions are provided to assure correct equipment placement. Services to the vessels are provided by jumpers between the nozzles in the service plugs and nozzles on the vessels. The location of all nozzles is known to ±1/16 in. in the X, Y and Z coordinates. Crane-mounted impact wrenches are used to operate the connector heads. Special lifting bails are used to handle the various equipment pieces with the cell crane.

Master-slave manipulators supplement the overhead crane and permit remote handling while performing maintenance on portions of the spray calciner/in-can melter and the high-level waste canister lid-weld/leak-test station. A remotely operated electro-mechanical manipulator is available to perform remote manipulations that cannot be performed with the master-slave manipulators.

6.3 EQUIPMENT LIFE AND SPARE REQUIREMENTS

Table 6.1 presents a preliminary analysis of expected equipment performance for major processing equipment and for selected auxiliaries. The principal failure modes have been identified and the anticipated service life of failure-prone equipment has been estimated. Typical problem areas are equipment plugging, motor failure due to failure of insulation as a result of high radiation and heat, electrical heater failure due to solids deposition, and reboiler leakage due to corrosion. The most common failures are auxiliaries such as pumps, agitators, valves, flowmeters and temperature elements. If the equipment is designed appropriately, replacement of these auxiliaries can usually be achieved by simple jumper changes in one to three shifts. Assuming replacement components are on hand, new jumpers can usually be fabricated within two to three days.

In case of a vessel failure, the vessel can either be removed from the equipment module after disconnecting the associated jumpers, or the complete module can be removed and replaced remotely after removing all of the connecting jumpers. Replacement of an entire module including disconnecting and reconnecting jumpers may involve several days of operating downtime.

To avoid undue plant downtime, a spare equipment inventory must be maintained or the equipment must be readily available from suppliers. This is especially true for specialized equipment that has a long lead time. Table 6.1 also gives a preliminary recommendation of the number of equipment spares that should be provided. In general, the design philosophy calls for vessel standardization wherever possible to permit the use of common spares and thereby reduce the cost of the spare parts inventory.

Basket-type centrifuges, which have been used extensively at the Hanford Purex and B-Plants in fission-product processing, are subject to motor and bearing failure due to heat and radiation. Two spare centrifuges are suggested

TABLE 6.1. Failure Analysis and Recommended Spares

System	Equipment Item		Principal Failure Mode	Expected Service Life	Maintenance Action	Number of Spares Recommended
A	CE-A-1	Primary centrifuge	Motor/bearings/plough failure	6 months	Replace centrifuge	2[a]
A	CE-A-2	Secondary centrifuge	Motor/bearings/plough failure	1 year	Replace centrifuge	2[a]
B	F-B-1	Primary sand filter	Solids plugging	1 year	Replace	1[a]
B	F-B-2	Polishing sand filter	Solids plugging	1 year	Replace	1[a]
C	IEX-C-1	Cs ion-exchange column	Resin plugging	2 years	Replace	1[a]
C	IEX-C-2	Cs ion-exchange column	Resin plugging	2 years	Replace	1[a]
C	F-C-1	Zeolite fixation bed	Resin plugging	1 year	Replace	1
D	IEX-D-1	Sr ion-exchange column	Resin plugging	2 years	Replace	1
D	IEX-D-2	Actinide ion-exchange column	Bed plugging	2 years	Replace	1
E	CA-E-1	Spray calciner furnace	Heater failure	1 year	Replace	1
E	CA-E	Spray calciner barrel	Corrosion--cracking	1 year	Replace	1
E	F-E-1	Calciner filter	Cartridge--plugging, cracking	6 months	Replace	1
E	ICM-E-1	In-can melter furnace	Heater failure	6 months	Replace	2
E	ICM-E-2	In-can melter furnace	Heater failure	6 months	Replace	2[a]
E	ICM	Connecting assembly	Solids plugging	1 year	Replace	1
E		Canister lid welder	Motor failure	1 year	Replace	1
E		Canister decontamination spray rings	Nozzle plugging	1 year	Replace	1

(a) Common spares.

over the life of the project. Failures of the in-can melter furnaces are anticipated due to electrical heater failures. Two spare melter furnaces are suggested. The spray calciner filters could also be a high-failure item. Two spare filter assemblies are recommended.

In addition to the spare equipment listed in Table 6.1, a complete inventory of auxiliary equipment consisting of pumps, agitators, connectors, jets, spray calciner vibrators, etc., should also be on hand. It may also be prudent to have one spare of each of the standardized-type vessels.

7.0 PROCESS FACILITIES

This section briefly describes the WNYNSC facilities to be used for the chemical treatment and conversion of the high-level waste to a solid, and discusses the modifications, installation procedures and equipment arrangements for the reference process.

7.1 DESCRIPTION OF UTILIZED FACILITIES

The exterior appearance of the fuel reprocessing plant at the WNYNSC resembles that of a small chemical plant. The major structure, shown in Figures 7.1, 7.2, and 7.3 (drawing SK-3-22817)(a), is the main processing building (27 m high, with a ventilation stack rising 61 m above the ground). This building is a complex structure made up of many cells in which spent nuclear reactor fuel was reprocessed using the Purex process. The building also contains fuel receiving and storage facilities.

Most of the high-level waste processing equipment (Systems A through G; see Section 6.0) is located in the Chemical Processing Cell. The final salt solution concentration and packaging equipment (System H) is located in the Scrap Removal Room. Portions of the final process effluent cleanup equipment are located in the new (existing 01-14 Building) unused Process Ventilation Cell (see Section 9.2.1).

7.1.1 Chemical Processing Cell

The Chemical Processing Cell is 6.7 m x 28 m x 13.1 m high and has a wall thickness of 1.8 m of ordinary concrete. The cell is serviced by a two-hook (2 ton and 16 ton capacity) remotely operated crane and an electro-mechanical power manipulator. Viewing necessary for remote operation is provided by oil-filled shielded viewing windows in the west and north walls of the cell. A track-mounted cart is provided for remote movement of equipment from the Equipment Decontamination Room into the Chemical Processing Cell.

The functions formerly carried out in the Chemical Processing Cell included dissolving, feed preparation and sampling, waste evaporation and

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(a) All referenced SK drawings are included in Appendix A.

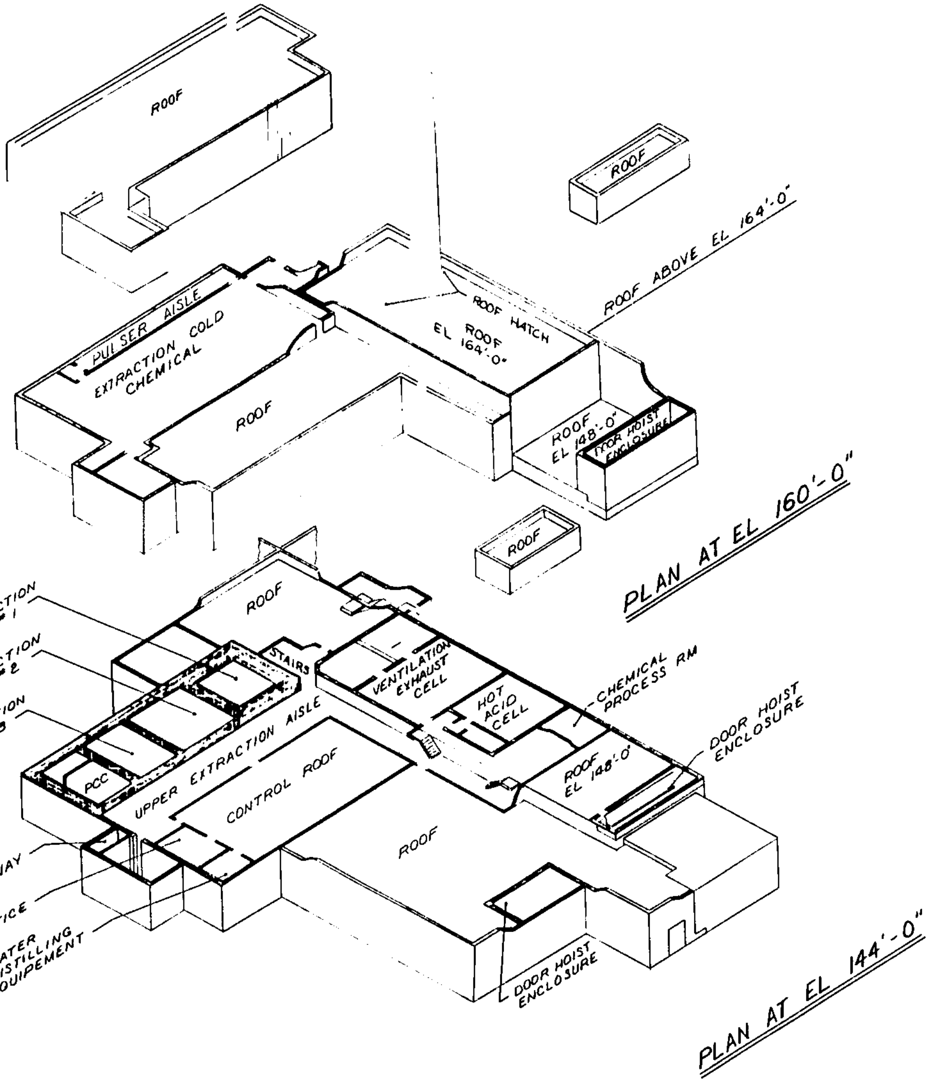

FIGURE 7.1. Major Features of the Upper Portion of the Main Processing Building

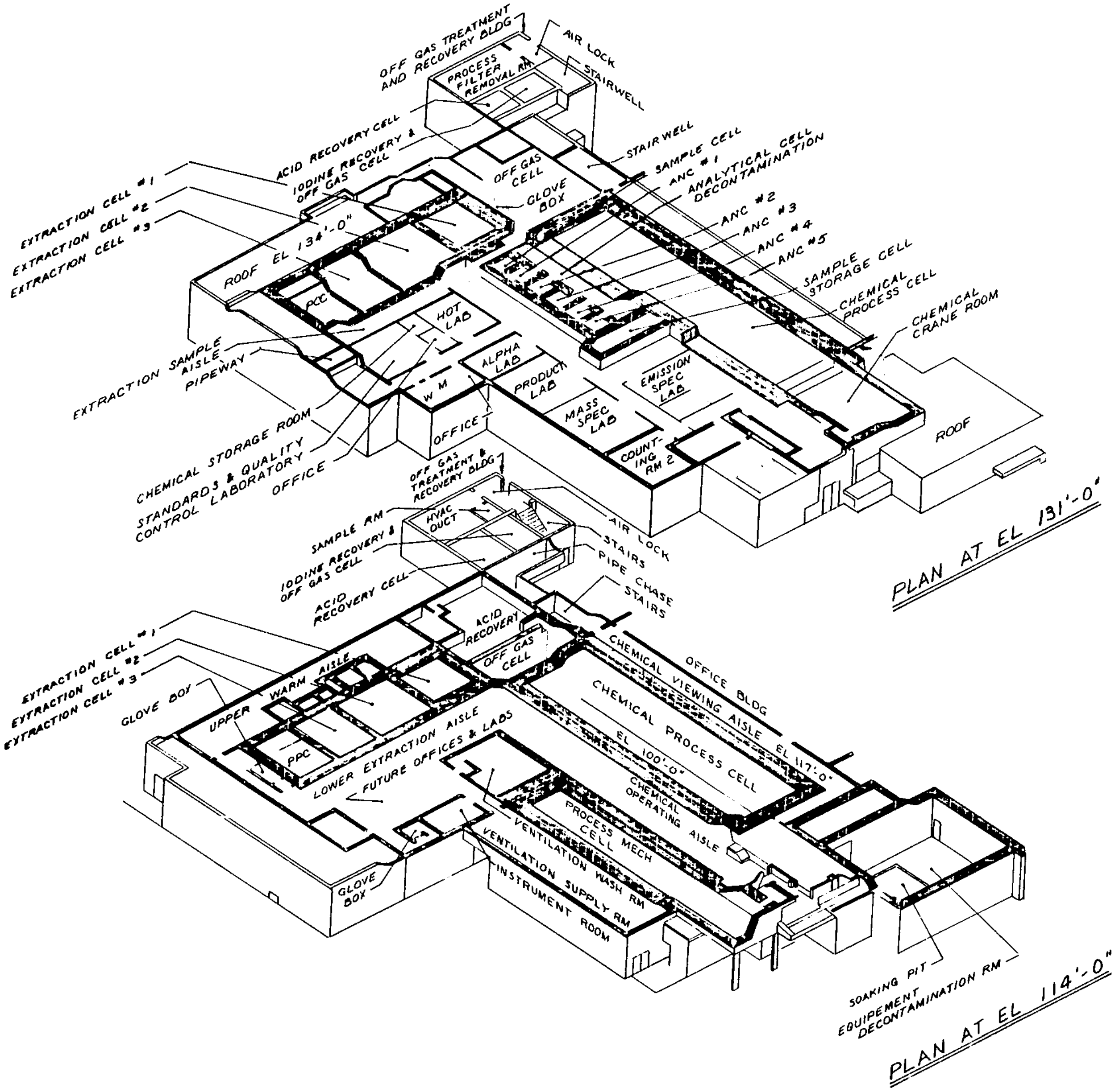

FIGURE 7.2. Major Features of the Middle Portion of the Main Processing Building

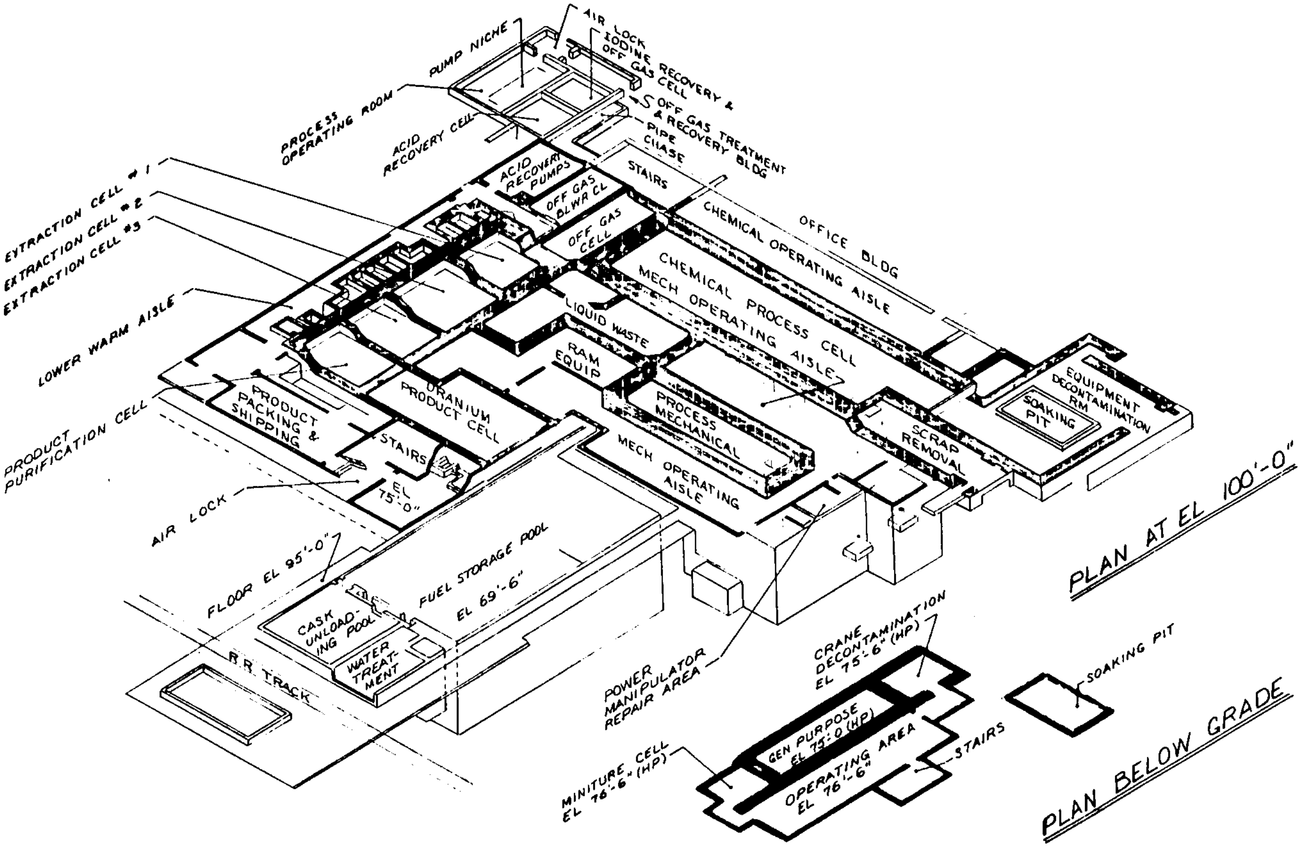

FIGURE 7.3. Major Features of the Lower Portion of the Main Processing Building

rework operations. The dissolvers are of an annular design and are installed around a solid plug of neutron-absorbing borated concrete. These will be removed before installing the waste processing equipment.

7.1.2 Scrap Removal Room

The Scrap Removal Room is 3.0 m x 11.6 m x 4.0 m high and has a wall thickness of 1.0 m of ordinary concrete. The room is serviced by a 7-1/2 ton remotely operated crane. Viewing is provided by one oil-filled shielded viewing window. The room has no installed equipment.

The functions formerly performed in this room were packaging and removal of fuel hardware and cladding residues from the chop-leach process.

7.1.3 Process Control Room

The existing Process Control Room will be used. Portions of the existing unused instrumentation will be removed to provide space for the instrumentation required for operating and monitoring the high-level waste processing equipment. A block diagram of the instruments required is presented in drawing SK-7-913.

7.1.4 Associated Operating Space

Various operating spaces adjacent to the Chemical Processing Cell and Scrap Removal Room will be used. Some modifications are required (see Section 7.3.3).

7.2 UTILIZATION OF EXISTING PROCESSING EQUIPMENT

For the most part it was concluded that it would be advantageous to provide new processing equipment rather than to attempt to use existing equipment. Not only is the existing equipment contaminated, but it is either not the right size and function or it is not compatible with the design required to fit the equipment into the available space. In addition, it is very difficult to analyze the reliability of used contaminated equipment.

7.2.1 Off-Gas Cleanup

New off-gas blowers and associated filters were provided in the new (01-04 Building), noncontaminated Process Ventilation Cell which was under construction by Nuclear Fuel Services as part of their planned expansion project at the time of their decision to terminate reprocessing operations. These items will be used as part of the process off-gas cleanup system. It will be necessary to complete the inlet piping from the Chemical Processing Cell and discharge piping to the stack.

7.2.2 Utility Service Requirements

The utility service requirements for the high-level waste processing system are shown in Table 7.1. The values shown include an excess capacity factor of 25%.

TABLE 7.1. Utility Service Requirements

Item	125% Load
Electrical power, kW	1,600
Steam, lb/h	8,400
Cooling water @ 35°C, gpm	630
Cooling water @ 25°C(a), gpm	310
Demineralized water, gpm	1.3
Compressed air, cfm	
Service	135
Instrument and control	100
Building services, Btu/h	
Chemical Process Cell heat load(b)	1,540,000
Scrap Removal Room heat load	158,000
Discharge streams	
To stack(c), cfm	80
To sump and drains, gpm	3

(a) Chiller provided as part of this preconceptual design study.
(b) Auxillary cell cooling being provided.
(c) From process vessels only.

7.3 PROCESS EQUIPMENT INSTALLATION

Installation of the waste processing and conversion equipment into Nuclear Fuel Services' reprocessing facilities will require some adjustments and modifications of those facilities. This subsection discusses and/or describes: 1) cell preparation and modifications, 2) installation procedures and 3) equipment arrangements.

7.3.1 Cell Preparation and Modifications

For the purpose of this preconceptual design study, it is assumed that all of the existing equipment has been removed from the Chemical Processing Cell and Scrap Removal Room and that the cells have been decontaminated down to radiation dose levels of approximately 10 to 12 mR and 1 mR, respectively. It is also assumed that although protective clothing is required for working in these areas respiratory protection (masks) will not be required except during welding.

Considerably more equipment is required for the high-level waste vitrification process than is currently installed in the Chemical Processing Cell. Consequently, many additional services will be required. Approximately 50 new penetrations, ranging in size from 2 in. in dia to 14 in. in dia, are estimated to be required for these services. (See detail 1, SK-7-967 for a typical service plug concept.) The locations for many of the service penetrations are shown on SK-7-961 and SK-7-962.

A shielded cubicle and a wall penetration sloped at a 45^{0} angle are required at the south end of the Chemical Processing Cell (see SK-7-962). Their purpose is to house the frit air-lock valves and the spray calciner filter blowback valves and to add frit to the spray calciner/in-can melter.

The existing equipment door in the north end of the Chemical Processing Cell will be used for moving both empty and full waste canisters into and out of the cell. A new transfer cart (see SK-7-966) will be required for this service. The existing rails through the door will be used. Two new shielded viewing windows will be required in the west wall of the Chemical Processing Cell for operation and maintenance of the spray calciner/in-can melter (see SK-7-961 and SK-7-963). Master-slave manipulators will be required at each of these windows and at one of the existing windows where canister welding and leak testing will be performed. Modifications to the Chemical Viewing Aisle (see SK-7-959) will be required to accommodate the new windows and to provide an operating room for the in-can melters. Various electrical switch gear and instrumentation will be housed in this room. Several through-wall periscopes will be required for viewing remote operations at the upper elevations of the Chemical Processing Cell.

A closed-loop refrigerated cooling system will be installed in the Chemical Processing Cell to remove the excess process heat resulting from the additional equipment needed for the vitrification process.

Various modifications, new tanks, etc., including a room at the 144-ft elevation to house the frit feeder, are required for aqueous makeup and chemical additions to the process equipment. These modifications are shown on SK-7-958.

One additional shielded viewing window complete with master-slave manipulators is required in the Scrap Removal Room. Modifications to the outside door of the Scrap Removal Room are required to provide for moving drums of salt out of the cell. While not part of this design effort, an air lock outside of this door may be required for contamination and/or air balance control.

7.3.2 Installation Procedures

The equipment that is to be installed in the Chemical Processing Cell is designed for complete remote operation, maintenance and removal; however, it will be necessary to enter the cell to install the equipment.

A modular design approach was used for the Chemical Processing Cell equipment. After the cell has been decontaminated, positioning trunions, dowels, etc., will be installed in the cell. The equipment will then be brought into the cell and placed on the positioning devices. After placing the equipment modules, precise measurements will be taken to locate all equipment nozzles. These measurements will be used for fabrication of the connecting jumpers. After contact fit-up, installation, and remote checkout of the jumpers, the as-built measurements will be entered on the appropriate drawings.

The equipment that is to be installed in the Scrap Removal Room is designed for contact installation and removal. However, the equipment is designed for remote operation and limited remote maintenance.

7.3.3 Equipment Arrangement

One of the objectives of this preconceptual design effort was to determine the feasibility of installing and operating the reference process equipment in existing cells at the WNYNSC. To accomplish this, fairly extensive equipment arrangement drawings were prepared. The arrangement of the equipment that is to be located in the Chemical Processing Cell is shown on drawings SK-7-961 through SK-7-967 and SK-7-980 and SK-7-981. The arrangement of the equipment that is to be located in the Scrap Removal Room is shown on drawings SK-7-969 and SK-7-970. The arrangement of the associated operating aisles is shown on drawings SK-7-958, SK-7-959 and SK-7-960. After preparation and review of the equipment arrangement drawings, it was determined that it is indeed feasible to accomplish the vitrification of the high-level waste with the spray calciner/in-can melter process in the existing facilities.

7.3.4 Alternative Equipment Installation

It was noted in Section 3.1.4 that the liquid-fed ceramic melter could be an advantageous replacement for the spray calciner/in-can melter. To assure that the West Valley preconceptual design would be compatible with the advancing technology a cursory evaluation was made to identify the flowsheet and facility modifications that would be required to accommodate the liquid-fed ceramic melter. This preliminary evaluation assumed that the flowsheet and facility design would be primarily based on the spray calciner/in-can melter process.

Process Modifications

Some process modifications will be required for replacing the spray calciner/in-can melter system with a liquid-fed ceramic melter. Process differences are in the: 1) vitrification feed system, 2) gaseous effluent treatment system, and 3) solidification process service requirements.

In liquid-fed ceramic melter process operations, the glass formers are blended with the nuclear waste and the mixture is fed to the process. The glass-former/waste feed mixture in the blend tank will require more efficient agitation. This can probably be achieved by a pulsed tank concept, which has been used by the Germans. An air-lift system feeds the blended glass-former/

waste mixture to the liquid-fed ceramic melter. Several air-lift stages are needed to achieve the required height in the Chemical Processing Cell.

The effluent system designed for the spray calciner's atomizing air is oversized for the requirements of the air flow from the liquid-fed ceramic melter which consists primarily of steam and inleakage air. However, a large-capacity effluent treatment system for the liquid-fed ceramic melter is desirable to accommodate unexpected pressure surges. The appropriate capacity of the effluent treatment equipment would need to be determined. However, the same equipment pieces are appropriate.

The liquid-fed ceramic melter process requires cooling water, which is not required for the spray calciner/in-can melter process. This cooling water cools the melter containment vessel, thereby contributing to the prevention of glass leakage into the refractory and thus increasing the melter lifetime. Depending upon the specific melter design, additional power may be required to operate the liquid-fed ceramic melter system.

The glass production rate required for the liquid-fed ceramic melter has been demonstrated in melters operated in nonradioactive development laboratories (Buelt and Chapman 1979). There are, however, substantial equipment developments required to implement this process at the WNYNSC (Holton 1981).

Facility Modification Requirements

A conceptual arrangement for the liquid-fed ceramic melter in the normal spray calciner/in-can melter location is shown on SK-3-22823, sheets 1 and 2. From the brief evaluation it appears that the only anticipatory facility modification that would be required is the installation of a pit directly below the location of the spray calciner/in-can melter. The pit would be required to provide sufficient head room under the liquid-fed ceramic melter for 10-ft-long canisters. The elevation of the liquid-fed ceramic melter installation is dictated by the location of the viewing windows and the master-slave manipulators, which are both necessary for operation of the unit.

8.0 SCHEDULE

8.1 ASSUMPTIONS

An operating contractor will be selected by the Department of Energy. This contractor will perform a review of the preconceptual design and will further develop the conceptual design with the assistance of various the Department of Energy laboratories.

8.2 ENGINEERING

An Architect Engineer (A-E) will be selected by the Department of Energy with the assistance of the operating contractor for Title I and II Design and Title III Engineering Services.

Title I and Title II design for the reference process for the chemical treatment and vitrification of high-level waste should be completed in 30 months. This 30-month design schedule will require early agreement on the conceptual design by all project participants to firmly establish the reference design.

8.3 CONSTRUCTION

The schedule discussed below assumes that the operating contractor will advertise for bids and award a fixed-price construction contract for the projects. The exact methodology for procurement and fabrication in connection with the construction is to be determined.

The construction bid and award period is estimated to be six months. The construction is estimated to take 18 months. This schedule is based on the assumption that the Chemical Processing Cell, Scrap Removal Room, Process Chemical Room and the Chemical Operating Aisle will have been cleaned out and decontaminated before starting construction of the reference process. The schedule also assumes early funding and implementation of supporting research and development activities, including verification studies of the reference process with WNYNSC waste.

8.4 STARTUP

A six-month startup phase should be adequate for this project.

9.0 COST ESTIMATES

9.1 EQUIPMENT AND CONSTRUCTION COSTS

An integrated-systems approach was adopted in estimating the equipment and construction costs rather than the usual approach of estimating by disciplines, such as mechanical, piping, electrical, instruments, etc. In this approach, the costs are estimated by systems such as System A (Feed Centrifugation), System B (Agglomeration, Settling and Filtration), etc. The specific system process and instrument diagrams, vessel drawings, cell arrangement drawing, and data sheets served as inputs for each system estimate. The advantage of the system approach is the identification of unit operations and equipment items that may have an undue impact on the overall project cost.

The units included in each system are shown on the Process Key Diagram, SK-7-901. The estimate also has sections for:

- general requirements
- facility modifications and miscellaneous equipment
- modifications to the Chemical Processing Cell and Scrap Removal Room
- system aqueous makeup (cold chemical systems).

The detailed estimate sheets show each item of equipment and the services related to that item as one line on the estimate. The detailed cost estimate, including the breakdowns of this estimating method, is presented in Appendix C. Each estimated item includes all appropriate costs that are related to and can be identified with the unit being costed. Lines, jumpers, supports, etc., are listed as part of the individual unit only as long as they can be identified with this particular piece of the system. Beyond this, costs are shown as part of the system "Common Area" or in the general sections listed previously. The preconceptual cost estimate, in first quarter 1981 dollars, is $28.6 million. A breakdown of this cost estimate is presented in Table 9.1.

TABLE 9.1. Preconceptual Construction Cost Estimate--West Valley Demonstration Project Waste Processing and Conversion

Description	Labor	Material	Equipment	Subcontractor	Totals	Contingency (37%)	Totals
General requirements	1,038,300	106,500	54,000	73,600	1,272,400	470,800	1,743,200
System A	228,400	275,700	590,500		1,094,600	405,000	1,499,600
System B	243,600	314,800	297,900		856,300	316,800	1,173,100
System C	371,500	493,200	498,900		1,363,600	504,500	1,868,100
System D	126,700	171,000	184,600		482,300	178,500	660,800
System E	245,700	398,300	724,000		1,368,000	506,200	1,874,200
System F	217,300	297,700	331,400		846,400	313,200	1,159,600
System G	129,900	200,200	148,300		478,400	177,000	655,400
System H	166,100	286,300	93,000		545,400	201,800	747,200
Aqueous Makeup	104,800	62,100	13,200	2,100	182,200	67,400	249,600
Facility Mod. and Misc. Equip.	157,400	70,300	1,195,500	4,474,400	5,897,600	2,182,100	8,079,700
Mod. to CPC and scrap removal room	369,600	458,700	99,600		927,900	343,300	1,271,200
Construction Cost Totals	3,399,300	3,134,800	4,230,900	4,550,100	15,315,100	5,666,600	20,981,700
Engineering					4,800,000	1,536,000	6,336,000
Project Management					918,900	340,000	1,258,900
Round-Up						23,400	23,400
Totals					21,034,000	7,566,000	28,600,000

The usual project adders for temporary construction, fire protection, project supervision, facility modification, contingency, etc., were applied to the summary cost estimate.

9.2 BASES FOR CONSTRUCTION COSTS

9.2.1 Radiation Levels

The estimate is based on the performance of all cell modification work at 10- to 12-mR/hour radiation exposures. It is assumed that the bulk of work will be performed in shops, galleries, and nonradioactive areas and that in-cell work will be confined to such activities as installing trunnions for the subsequent remote positioning of equipment modules and jumpers, anchoring of new piping modules, etc. Concrete coring for the new piping modules would be performed from the relatively "cold" service gallery side.

With the Department of Energy's current personnel radiation limits of 300 mR/week, 3 R/quarter and 5 R/year and the anticipated availability of craft personnel, craft exposure is not expected to be a significant problem at the assumed 10- to 12-mR/hour rate. However, workers who receive a 300-mR/week exposure in approximately a 30-hour period would be paid for 40 hours work. Required dress in this area would be white coveralls, booties, and gloves, but masks and/or fresh air supply would not be required. However, masks would be required to weld to existing items in the cells.

The cost of five full-time radiation monitors for the duration of the construction period is included in the estimate.

9.2.2 Hourly Rates

The labor rates used are those currently in effect in the Buffalo area and include fringe benefits, labor insurance, and taxes. No craft overtime was included. A 12% factor was added to allow for travel. A job factor of 10% of labor was used due to the unfamiliarity of the crafts working in a radiation zone and the need for special clothing.

9.2.3 Construction Workers

Although the availability of trained construction workers might be somewhat limited in the immediate West Valley, New York, area, a ready pool of qualified craft personnel is believed to be available in the Buffalo and Jamestown, New York, areas.

9.2.4 Earthquake

A Uniform Building Code Zone 2 earthquake requirement was used in the estimate for bracing considerations. At this stage of the design, the bracing is not shown but allowances were made in the estimate increased material and time requirements.

9.2.5 Escalation

Escalation was not included in this estimate. An overall project schedule for the West Valley Demonstration Project will be developed. All cost elements for the project will be escalated to the scheduled performance period at a later date.

9.2.6 Contingency

The status of equipment availability is broken into four categories as follows:

category 1--off the shelf, item used as is, no modification
category 2--off the shelf with modifications
category 3--custom design and fabrication
category 4--research and development, custom design and fabrication.

Contingencies applied to category 1, 2, 3, and 4 equipment were 20%, 30%, 40%, and 50%, respectively. From these values, a weighted overall contingency (see Table 9.1) of 37% was developed for the entire project.

9.2.7 Engineering

The total cost of engineering design (Title I, Title II, and liaison during construction) was developed by Vitro Engineering Corporation and is 22% of the construction cost. This relatively high ratio of engineering to construction is typical for chemical separations facilities where significant radioactivity is present and remote maintenance for facility and equipment designs is required.

Complete detail designs of vessels and piping are required for the Chemical Processing Cell owing to the highly specialized nature of the equipment and the need for locating all vessel and service nozzles within a ± 1/16-in. tolerance in the X, Y and Z coordinates.

9.3 EXCLUSIONS FROM THE EQUIPMENT AND CONSTRUCTION COST ESTIMATE

The following items were excluded from the estimate:

1. startup costs
2. interest on funds during construction
3. cell decontamination
4. research and development testing
5. storage facility for filled canisters
6. escalation
7. updating of services such as steam, compressed air, cooling water, electric power, etc., or increasing the capacity of services, except the facility transformer
8. costs for final cell and equipment decontamination and dismantling after completion of the waste vitrification program
9. craft overtime pay
10. operating costs.

9.4 ESTIMATE OF MANPOWER REQUIREMENTS

The process equipment design is based on a three-year operating campaign at a 60% on-stream factor and assumes a rotating four-shift, seven-day/week operation. A summary of the manpower requirements for the high-level waste process operation is shown in Table 9.2.

9.4.1 Direct Process Operator Requirements

A combination of continuous and batch processing is used. Table 9.3 provides a preliminary estimate of direct process operator requirements for

TABLE 9.2. Summary of Manpower Requirements for High-Level Waste Processing

Personnel	From Table	Manpower Normal(a)	Manpower Plant Startup
Direct process operators	11.4	27.0	50.0
Supporting shift manpower	11.5	84.0	136.0
Miscellaneous supporting manpower	11.6	34.0	44.0
TOTAL		145.0	230.0

(a) After plant shakedown period.

TABLE 9.3. Direct Process Operator Requirements for Systems A Through H--High-Level Radioactive Waste Processing

Direct Process Operators	Operational Mode(a)	Operators Required/Shift Normal(b)	Operators Required/Shift Plant Startup	Total Operators Required/Day(c) Normal(b)	Total Operators Required/Day(c) Plant Startup
System A	B & C	0.5	1.0	2.0	4.0
System B	B & C	0.5	1.5	2.0	6.0
System C	B & C	2.0	3.0	8.0	12.0
System D	B	0.25	1.0	1.0	4.0
System E	B & C	2.0	3.0	8.0	12.0
System F	C	0.25	1.0	1.0	4.0
System G	C	0.25	0.5	1.0	2.0
System H	B & C	1.0	1.5	4.0	6.0
TOTAL		7.0	13.0	27.0	50.0

(a) B = batch and C = continuous.
(b) After plant shakedown period.
(c) Assumes four-shift, seven-day/week operation.

systems A through H for plant startup and normal operation. Personnel skilled in operating master-slave manipulators are required for Systems E and H. Also identified are the major operations that are performed in batches. Solids centrifugation, gravity settling, cesium and strontium ion exchange, zeolite fixation of cesium, high-level waste canning and drumming of decontaminated salt are examples of batch processing steps. Batch cycle times appear on the process flow diagrams (Appendix A) and in the tables in Section 5.3.

9.4.2 Support Personnel

Support personnel, as shown in Tables 9.4 and 9.5, are required for operation and maintenance of the high-level waste processing system. Skilled crane operators are required for handling waste containers in the Chemical Processing Cell during normal plant operations and for remote changeout of in-cell

TABLE 9.4. Supporting Shift Manpower Requirements--High-Level Radioactive Waste Processing

Support Personnel	Manpower/Shift Normal(a)	Manpower/Shift Plant Startup	Total Manpower/Day(b) Normal(a)	Total Manpower/Day(b) Plant Startup
Pipefitters	2.0	3.0	8.0	12.0
Millwrights	1.0	3.0	4.0	12.0
Instrument technicians	2.0	4.0	8.0	16.0
Crane operators	1.0	1.0	4.0	4.0
Electricians	2.0	4.0	8.0	16.0
Utility operators	2.0	2.0	8.0	8.0
Radiation monitors	2.0	3.0	8.0	12.0
Operation shift supervisor(c)	1.0	1.0	4.0	4.0
Process engineers	2.0	4.0	8.0	16.0
Analytical sampling	2.0	4.0	8.0	16.0
Analytical chemist	2.0	3.0	8.0	12.0
Analytical counting	1.0	1.0	4.0	4.0
Analytical shift supervisor(c)	1.0	1.0	4.0	4.0
TOTAL	21.0	34.0	84.0	136.0

(a) After plant shakedown period.
(b) Assumes four-shift, seven-day/week operation.
(c) Possibly has additional duties.

equipment and jumpers. A technical staff consisting of analytical chemists, health-physics, process technology and environmental personnel is required to provide support to operations staff. Service personnel are required for maintaining the process equipment and for fabricating replacement parts. It may be prudent to contract for a major portion of the service work with an outside firm and keep only key service personnel.

TABLE 9.5. Miscellaneous Supporting Manpower Requirements--High-Level Waste Processing

	Manpower	
Support Personnel	Normal	Plant Startup
Relief crane operator	1.0	1.0
Relief/day shift process operators	4.0	6.0
Relief/day shift utility operators	4.0	6.0
Relief/day shift radiation monitors	2.0	2.0
Relief/day operating supervisor(a)	1.0	1.0
Day shift operating supervisor(a)	1.0	1.0
Relief analytical personnel	2.0	2.0
Relief/day shift process engineers	2.0	3.0
Industrial engineer	1.0	2.0
Environmental engineers	1.0	2.0
Relief/day shift pipefitters	1.0	1.0
Relief/day millwrights	3.0	3.0
Relief/day instrument technicians	1.0	1.0
Relief/day electricians	1.0	1.0
Laborer	1.0	2.0
Carpenter	1.0	2.0
Painter	1.0	1.0
Welders	3.0	4.0
Maintenance scheduler(a)	1.0	1.0
Maintenance expediter(a)	1.0	1.0
Maintenance supervisor(a)	1.0	1.0
TOTAL	34.0	44.0

(a) Possibly has additional duties.

10.0 RESEARCH AND DEVELOPMENT REQUIREMENTS

The process flowsheet and equipment definitions for the preconceptual design study are based upon existing technology that has been developed in Department of Energy laboratories and is being used in nuclear waste management facilities operated at defense sites. The technical data developed at these sites are liberally used to define the reference solidification process flowsheet and material balances.

There are significant physical and chemical differences between the WNYNSC high-level wastes and the high-level wastes stored at the government-operated defense sites. The actinide and fission-product contents of the WNYNSC high-level wastes are ten to one hundred times higher than those of the defense wastes. These differences are due primarily to the higher burnup fuel reprocessed at the WNYNSC site and the nature and efficiency of radionuclide recovery during fuel reprocessing operations. Because of these differences, the technical data developed at the defense waste management sites may not be strictly applicable to management of the WNYNSC high-level wastes. Higher radiochemical decontamination efficiencies will be required for the salt fraction of the WNYNSC waste than for the salt fraction of defense waste to minimize worker exposure and to produce an acceptable low-level salt cake.

A principal feature of the salt/sludge separation process is the physical and chemical separation of the radioactive waste constituents from the bulk of the nonradioactive waste constituents. The higher radionuclide levels of the WNYNSC wastes stretch the technical capability of the processes to achieve an adequate separation of radioactive constituents from the nonradioactive components. Most of the salt/sludge separation process should be verified experimentally using the actual waste.

The major research and development requirements for the salt/sludge separation process, as outlined below, consist of 1) verification of the salt/sludge separation flowsheet and 2) process improvements to simplify the process, to make it reliable, or to improve flowsheet performance.

10.1 VERIFICATION OF THE FLOWSHEET

The development efforts needed to support the existing salt/sludge separation flowsheet consist of these general areas:

- physical separation processes
- ion-exchange processes
- waste form development.

10.1.1 Physical Separation Processes

The reference preconceptual flowsheet includes the physical separation processes of centrifugation, gravity settling, and filtration. These three unit operations remove essentially all of the insoluble radiochemical activity from the supernate.

As a general rule in the chemical process industry, these unit operations are experimentally evaluated using actual plant feedstock materials. Initial development work using simulated feed material provides a basis for process equipment selection. Final verification of the process equipment performance using actual radioactive solutions is impractical on a plant scale. However, laboratory-scale tests can be conducted to predict equipment performance.

Physical data, such as viscosities and densities, are necessary for plant design and can be obtained in plant-scale nonradioactive tests. Simulated nonradioactive streams can also be used to determine operating performances of solid/supernate separations. The operating parameters that need to be measured for the salt/sludge separation process include:

- for the centrifuge--
    - estimates of cycle times
    - loading rates
    - basket volumes
    - separation efficiencies
    - identification of required remote design features
- for the gravity settlers--
    - solids settling rates

need and effectiveness of coagulating agents
variables to achieve efficient separations, e.g., thermal cycling, solid recycling

- for the sand filters--
  efficiency of last solids removal
  pressure drop
  loading characteristics
  ease of bed regeneration.

10.1.2 Ion-Exchange Separations

The ion-exchange process is sensitive to chemical variations in the feed systems. Factors such as the Na/Cs ratio, salt concentration, pH, presence of complexing agents, ion-exchange poisons, resin integrity, ion-exchange kinetics, elution kinetics and regeneration procedures need to be assessed using WNYNSC waste. The effect of temperature on the efficiency of the process is important since the ion-exchange units are located in a cell that will be thermally hot. Initial nonradioactive development scope studies can assess the entire ion-exchange process. Laboratory-scale radioactive tests can verify process performance. The ion-exchange characteristics for the soluble actinides (Am, Cm and Pu) are important and need to be determined for all ion-exchange columns in the system.

10.1.3 Waste Form Development

If it is necessary to prevent thoria phase separation, glass development studies need to be directed towards immobilization of the Thorex waste (assuming borosilicate glass to be final waste form). Thoria has a low solubility in borosilicate glasses ($\sim$10 wt%) and can limit the effective Thorex waste loading in the glass product. Any adverse effects on thoria phase separation need to be identified. Higher Thorex waste loadings in the final waste form product will minimize the volume of high-level waste produced and thus decrease the cost of the West Valley Demonstration Project.

An alternative to placing the thoria in glass is to separate and decontaminate the thorium from the acid waste. The volume of material to be put in glass is thus substantially reduced. The decontaminated thorium could be stored or disposed of separately. Additional processes and equipment would be needed for this step. On the otherhand, thorium could be separated by solvent extraction in the existing plant equipment prior to dismantlement of the processing equipment now in the Chemical Processing Cell.

10.2 PROCESS IMPROVEMENTS

The process flowsheet for Pacific Northwest Laboratory's preconceptual design study was based upon existing technology. Process improvements and alternative unit processes could result in simpler operations, enhance unit process performance, and improve process reliability. Potential improvements to the processes include ion-exchange and solid/liquid separation processes.

10.2.1 Ion-Exchange Process Improvements

The reference ion-exchange process for cesium removal uses a Duolite ARC-359 cation ion exchanger. The elution agent, NH_4OH-NH_2CO_3 solution, is displaced from the ion-exchange column, is separated from the cesium by steam stripping and is recovered for recycle. This process scheme adds several process vessels which, if not operated correctly, will not operate reliably (Richardson 1969). The process also requires a separate vent system for ammonia losses.

The use of another Duolite resin, CS-100, is under development at Savannah River Laboratory (Wallace and Ferguson 1980). The cesium can be eluted from this resin with formic acid; thus, elutriant recovery is not needed. Successful development of this process would permit simplifying the process flowsheet and plant operations. Another potential improvement in the ion-exchange processes for treating West Valley alkaline supernate is the application of improved ion exchangers to increase radionuclide removal. This may be necessary to produce a nontransuranic salt cake product. The improved removal may be obtained by either mixed-bed ion exchangers that contain several ion-exchanger materials selective to both actinide and fission-product cations, or by titanate or other mineral ion exchangers.

10.2.2 Physical Separations

Different process equipment pieces could be used to obtain the same effective separation as is achieved with the reference approach using centrifugation, gravity settling, and filtration. Two concepts that could reduce the overall equipment requirements are the pressure leaf filter and the inertial filter. These two approaches could reduce the cell floor space requirements. The leaf filter concept is a standard filtration approach used in the chemical industry and in low-level radioactive waste management systems in Europe. This filter may be implemented in high-radiation fields with minor modifications.

The inertial filter concept, developed by Mott, can provide the required solid/liquid separation, can potentially reduce the floor space and can be implemented in remote service. This system would not need the addition of a filter aid. This system has been used in nonradioactive tests at Hanford to evaluate separation of alkaline sludge and supernate.

Gravity settling alone can, potentially, replace the centrifuge steps. However, to maintain the same throughput approximately three times more floor area will be required. This approach can be used for the initial sludge/supernate separation process and for the sludge washing steps. However, approximately six wash steps will be required as opposed to three wash steps for the centrifuge wash operation. The possibility exists for implementing these processes in other cells at the facility, e.g., the Process Mechanical Cell.

10.2.3 Waste Form Process Improvements

The salt cake product produced from solidification of the decontaminated supernate is hydroscopic. Acceptable methods to dispose of this waste form, or additives that would improve the product quality, need to be assessed.

The wiped-film evaporator needs basic engineering development to be a reliable equipment piece in a remote environment. At the present time the wiped-film evaporator is the best equipment piece to be used for supernate solidification operations.

Alternative high-level waste vitrification processes, such as the liquid-fed ceramic melter, may become more accepted than the spray calciner/in-can melter process because of process flexibility and potentially improved waste form properties. Implementation of the liquid-fed ceramic melter process within the flowsheet needs to be evaluated.

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