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ATLANTIC RICHFIELD HANFORD COMPANY

QUARTERLY REPORT

PROCESS TECHNOLOGY AND PROCESS DEVELOPMENT

OCTOBER 1976 THROUGH DECEMBER 1976

Research and Engineering Division

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Richland, Washington 99352

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ABSTRACT

This quarterly report is the third in a series intended to provide information on research and engineering activities being performed to improve the processing of irradiated reactor fuels, the production of plutonium, and the management of resultant chemical wastes. The work reported here was performed during the period October through December 1976.

ATLANTIC RICHFIELD HANFORD COMPANY
QUARTERLY REPORT
PROCESS TECHNOLOGY AND PROCESS DEVELOPMENT

INTRODUCTION

This report describes the scientific investigations and engineering application programs being sponsored by the Energy Research and Development Administration (ERDA) under the direction of the Research and Engineering Division of the Atlantic Richfield Hanford Company (ARHCO).

This report presents summaries of technical progress. Brief statements of objective and prior work introduce individual portions of the report and appear in a type style different from that used to describe progress during the report period. The ERDA Division of Environmental Control Technology sponsors the mission entitled "Management of Surplus Facilities and Equipment;" other missions are sponsored by the Division of Nuclear Fuel Cycle and Production.

Previous periodical reports covering segments of this report's subject material were published as ARH-ST-110 A through D, ARH-ST-132 A through C, ARH-ST-128 A and B, ARH-ST-139, ARH-LD-111 A through E, and ARH-LD-149 A and B.

1.0 WASTE MANAGEMENT (COMMERCIAL)

The overall objective of the Commercial Waste Management mission is to develop and demonstrate the necessary technology and capability to safely maintain custody of commercial radioactive wastes so that they are not a threat to man and his environment now or in the future. The Commercial Waste Management program covers a broad spectrum of radioactive wastes generated by facilities such as fuel fabrication plants, fuel reprocessing plants, and nuclear power reactors. It includes such processes as waste treatment, packaging, shipping, and interim and final storage.

1.1 NATIONAL WASTE TERMINAL STORAGE PROGRAM

In February 1976, the Energy Research and Development Administration announced a greatly expanded waste management program for defense and commercial radioactive waste. In that announcement, ERDA indicated that an Office of Waste Isolation (OWI) would be created within Union Carbide Corporation - Nuclear Division (UCC-ND) with the responsibility for program management of the expanded program for the geologic disposal of commercially generated radioactive waste. This overall program is known as the National Waste Terminal Storage Program.

Atlantic Richfield Hanford Company has been assigned the responsibility for two elements of the program: (1) basalt studies - an assessment of the feasibility of storing radioactive wastes in the Columbia River basalts; and (2) Waste Containerization Studies - development of design criteria for the spectrum of canisters and packages required to contain the radioactive wastes that must be stored/disposed of at a Federal repository.

Columbia River Plateau Flood Basalt Studies - R. A. Deju
(R. E. Smith - Study Manager)

Information about the basalt must be obtained to allow a full assessment of the feasibility of using basalt as a terminal storage medium for radioactive waste while assuring environmental safety, permanent isolation, and initial retrievability.

During the quarter ending December 31, 1976, detailed plans were completed for all studies to be performed during fiscal year 1977. The plans defined work for the \$700,000 currently authorized, and an additional \$300,000 for recommended studies which could be performed during FY 1977. These plans were transmitted to Energy Research and Development Administration and the Office of Waste Isolation, Union Carbide Corporation, Nuclear Division.

Document ARH-ST-137, "Preliminary Feasibility Study on Storage of Radioactive Wastes in Columbia River Basalts," was printed and distributed. This document provides an extensive review of data obtained from past studies of the basalts and the report completes the documentation of preliminary, geological studies on Columbia River basalts at Hanford.

Data collection was initiated on the Sentinel Gap Geologic Study. Data on the Wallula Gap study is being compiled and a document is scheduled to be issued by June 1977.

Contracts have gone out for bids on the hydrologic studies and should be let by late January 1977.

A study of the upper interbeds intercalated between individual basalt flows has been completed and reviewed. It will be released after proper clearance in February 1977.

A contract to conduct compatibility studies and leaching experiments is being negotiated with Pennsylvania State University.

A talk entitled "Feasibility of Storing Nuclear Waste at Hanford" was presented at the Office of Waste Isolation Information Meeting in Oak Ridge, Tennessee, on December 8, 1976, by Raul A. Deju.

The feasibility of using geochemical techniques for flow identification was examined by Lawrence Berkeley Laboratory. Its report is presently being reviewed. Results confirm that the geochemical techniques are accurate.

Several visits to the U. S. Bureau of Reclamation construction site of the Second Bacon Siphon tunnel were made. We will continue to monitor their tunnel construction into the basalt.

Drilling of the ARH-DC-2 deep basalt well is being delayed until the completion of an Environmental Assessment and review of the Program by the appropriate State agencies. All other work to date is on schedule and their milestones during the quarter were met.

Waste Containerization Studies - D. C. Nelson, Program Manager

Two containerization studies, referred to as Task I and Task II, respectively, are the High-Level Waste Canister Envelope Study and the Radioactive Waste Packaging Study. The prime objective of Task I is to develop design criteria, specifications, and designs for the canisters that will contain the solidified high-level radioactive waste. The immediate objective of Task II is to develop input for a schedule and control program by identifying waste container problems (for all radioactive waste except high- and low-level wastes) and a plan by which these problems can be resolved. These studies were initiated during Transition fiscal year 1976.

Task I - High-Level Waste Canister Envelope Study -

J. R. LaRiviere

The program plan for Task I, High-Level Waste Canister Envelope Study, was developed and the final draft submitted to ERDA - Richland Operations Office, in November 1976.

The plan considers the canister to be part of an integrated handling and storage system which is the canister cycle from time of waste solidification to final waste disposition, including transportation. Since a number of processes and options for the waste management system have not yet been specified, alternatives will be studied at each stage in the canister cycle. The canister cycle includes: waste solidification, storage at the fuels reprocessing plant (FRP), shipment to geological storage, emplacement in geological storage and emplacement in final geologic storage.

Using the canister cycle as a base and functional canister performance criteria as a guide, the plan is divided into several activities. These activities will permit an orderly technology assessment and identification of areas where additional studies are needed to permit development of the canister design criteria. The activities include heat transfer, codes and standards, canister materials, transportation, handling and storage, and systems analysis.

During this quarter, Task I work continued on the technical assessment of the various program tasks, as outlined in the program plan. Contacts have been initiated and continued with personnel involved in the waste canister design work for both ERDA contractors and the commercial nuclear industry.

Purchasing action has been initiated for letting of contracts to obtain consultation services. Such services will consist of: heat transfer, preliminary engineering studies of waste shipping casks, codes and standards, materials, and structural analysis.

Task II - Radioactive Waste Packaging Study - H. A. Zweifel

The primary objective of this task is to develop design criteria for the spectrum of packages required to contain the commercially generated intermediate-level and low-level transuranic wastes that must be shipped to a Federal repository. The immediate objective is to develop a detailed study plan describing how these criteria are to be developed. This study plan is to be completed by January 31, 1977. Preliminary draft criteria are to be developed by January 31, 1978.

Since the start of this study, information has been reviewed on the status of waste packaging technology throughout the nuclear industry. Contacts have been made with personnel involved with radioactive waste packaging, handling, and storage. A preliminary program plan was developed and submitted to ERDA-RL in November 1976.

1.2 WASTE PROCESSING - COMMERCIAL, RESEARCH AND DEVELOPMENT

Remote Maintainability of Transuranic Waste Processing Equipment - K. H. Henry, D. C. Nelson

The objective of this program is to provide practical solutions to selected problem areas in the remote operation and maintenance of nuclear fuel cycle plants, and especially their radioactive waste handling and processing facilities. These solutions will be provided by establishing design bases through engineering studies, conceptual design, and hardware development and demonstration.

Progress on this new program during the first quarter of FY 1977 has been limited by the temporary assignment of key personnel to higher priority tasks. Initial onsite contacts have been made to begin the survey work, and plans for offsite visits are being formulated. Accelerated effort on this program during the second quarter of FY 1977 will be exerted to ensure that the initially established milestones are met.

1.3 SUPPORTING STUDIES - RESEARCH AND DEVELOPMENT

Sealed Storage Cask Demonstration - D. C. Nelson, Program Manager

The Federal government is required to assume permanent custody of all high-level radioactive waste within ten years of its generation by the commercial nuclear industry. Final waste storage sites and methods are under development on the National Waste Terminal Storage Program. In the event that these final storage facilities are not available when waste deliveries begin, above-ground storage in a Retrievable Surface Storage Facility (RSSF) may be required to provide interim storage until final storage facilities are established. Atlantic Richfield Hanford Company has studied various interim storage concepts and is currently conducting engineering and demonstration programs to verify the feasibility of the Sealed Storage Cask Concept (SSCC) - concrete-shielded steel casks placed outdoors and cooled by natural air circulation - as an interim storage concept. After completion of design verification in FY 1977, a demonstration unit containing radioactive waste will be put in service in FY 1978.

Conceptual Design Support Studies - J. M. Davis

A demonstration unit has been built and activities are being conducted to evaluate the integrity and feasibility of the SSCC. Important phases of this program are: (1) verification of heat transfer and radiation shielding calculations through tests with an SSC demonstration unit equipped with an electric heater to simulate decay heat from the waste, (2) evaluation of the structural integrity of the storage cask, (3) evaluation of canister decontamination techniques as a means of establishing acceptable contamination levels for the outer surfaces of waste-filled canisters received by the RSSF, (4) investigation of surveillance methods to assure waste containment during storage, (5) evaluation of canister and cask handling methods, and (6) fabrication of an SSC to be loaded with radioactive waste to demonstrate the feasibility of the concept. Phases 1 through 5 are currently in progress, while Phase 6 has not yet been initiated.

Heat Transfer Design Verification. The heat transfer design verification tests on the SSC demonstration unit were restarted again in late November, following successful heater repair and reassembly of the unit. By mid-December, thermal equilibrium was reached throughout the unit at 7.5 kw heater power, with all air inlet vents closed. The air outlet vents were then closed and equilibrium was reached by late December. The accumulated temperature data are being forwarded to Battelle-Northwest for analysis and comparison with calculated data.

Structural Evaluation. An interim report on the SSC was issued. The report summarizes all the various studies performed to select a final closure configuration for the SSC. The corner closure weld was ranked as best. Further structural analyses are being conducted on both the corner weld and the second-ranked threaded plug and seal weld.

Decontamination Evaluation. Battelle-Northwest is continuing their testing of Waste Solidification Engineering Prototype (WSEP) canisters in the mini wet basin equipment. Results are not yet available, but the basin water contamination levels appear to have come to equilibrium. The canister will be removed soon, smeared, and the data evaluated. The decontamination work proposed by BNW and described in the last Quarterly Report, will be initiated in January 1977.

Storage Area Surveillance. Hanford Engineering Development Laboratory (HEDL) has completed work on the design, fabrication, and testing of Geiger-Muller (GM) nuclear radiation detectors for surveillance of individual SSC integrity. The final report on this work was transmitted to ARHCO in December.

The goal of this activity was to develop a reliable design for a surveillance radiation detector for the SSC. The detectors should function unattended for long periods of time and should provide suitable signals to alert personnel to increases in detected radioactivity caused by a breach in an individual SSC within the facility.

Only partial success was achieved in the attainment of this goal. A prototype system was built and tested which could detect, and alarm with a 99 percent confidence, a 5.94 micro-

curie particle of ^{137}Cs located six inches from the detector, or an equivalent volume distribution of 3.4 nanocuries/cm in a 12-inch-wide air exit port. Reliability figures indicate a mean time between failure (MTBF), for the prototype which used inexpensive standard components, of about 8 years (excluding GM-tube failure rate). Additional work would be needed to strengthen the basic hardware of the prototype for field use, and methods for reducing the high background of radiation, within which the detector would operate, need to be developed. Limitations of time and money did not permit further studies for this additional work. Future work on radiation detectors will be directed specifically at the type of detection or surveillance instrumentation to be used on the radioactive demonstration.

Demonstration of Remote Welding and NDE. Work has continued on a task whose objective is to demonstrate that the storage cask can be remotely welded, using the corner weld closure design, and that the weld can be nondestructively examined using the ultrasonic technique developed by BNW.

Battelle-Northwest is currently developing a transducer support system and fabrication activities have been initiated for the welding torch angulation adaptor, rotator frame, thrust bearing, and mockup hot cell wall. Procurement action has been started on various items required for welder control.

Evaluation of Canister and Cask Handling. A program is being conducted to develop a grappling system capable of handling canisters and casks as they are processed in an SSCC facility.

Program objectives are to design, fabricate, and demonstrate a sealed container handling mechanism during FY 1977. A half-scale prototype sealed container handling mechanism is being fabricated. Testing of the mechanism will follow fabrication.

Decontamination and Decommissioning Technology for Nuclear Fuel Cycle Equipment and Facilities

The objective of this program is to develop and demonstrate improved technology for the decontamination and decommissioning (D&D) of radioactively contaminated equipment and facilities throughout the back end of the LWR fuel cycle.

The activity is composed of two tasks. Task I proposes to develop a high speed, reliable, and remotely operated metal cutting tool for cutting contaminated metal scrap of arbitrary shape and size to small, easy-to-handle pieces.

Task II proposes to identify initial design requirements to permit acceptable decommissioning of liquid water reactor (LWR) fuel reprocessing and recycle facilities at the end of their operating life.

This activity is now funded through the Savannah River (Light Water Reactor) Fuel Cycle Project Office.

Task I - Size Reduction - G. A. Beitel

Previous work by ARHCO on the Contaminated Equipment Volume Reduction (CEVR) program identified the arc saw (a spark erosion device) as the metal cutting tool offering the greatest promise of achieving the Task I objective stated above. Under that program, a procurement request to design and build an arc saw was issued. Furthermore, that program initiated the design of a test stand to be used for the arc

saw; the test stand design is in progress and will be completed under the CEVR Program. Essentially all other arc saw development work has now been transferred from the CEVR Program to this new program.

A bid was received from ReTech, Inc., of Ukiah, California, to design and build two arc saws, one with a 16-inch diameter blade, and the other with a 48-inch diameter blade. Both will be operated from a common power supply and control unit. The bid has been reviewed but further contract negotiations have been postponed until a vendor audit can be performed. It is hoped that the contract can be negotiated and design begun in March 1977.

Task II - Decommissioning - L. E. Kusler

Activities on this task were initiated during the first quarter of FY 1977 and included a review of:

- a. Documented experiences of decommissioning government facilities.
- b. ANSI Standard N300, "Design Criteria for Decommissioning of Nuclear Fuel Reprocessing Plants."
- c. Waste management and fuel cycle studies attempting to quantify the impact of decommissioning plants as part of the total waste management problem.

Data obtained from these reviews are being used to support Savannah River Laboratory in their conceptual design of Light Water Reactor Fuel reprocessing and recycle facilities. The initial work centers on the facilities required to process wastes generated by a 10 MTU/day fuel reprocessing plant.

Special StudiesGeneric Environmental Impact Statement - R. A. Kennedy

Atlantic Richfield Hanford Company has been participating in the preparation of the "Generic Environmental Impact Statement for Managing Radioactive Wastes from Commercial Reactors and Post-Fission Operations in the LWR Fuel Cycle (GEIS)." The document preparation is being coordinated by BNW with ARHCO preparing sections on the following subjects:

- o Failed Equipment and Noncombustible Scrap Treatment
- o Waste Solvent Treatment
- o Intermediate- and Low-Level Liquid Waste Treatments
- o Interim Storage of Solidified High-Level Waste at the Fuels Reprocessing Plant
- o Contingency Plan of an RSSF for Interim Storage.

Second drafts of the first three sections and first drafts of the other sections were transmitted to BNW in early December 1976. The primary purpose of the information submitted to date has been to provide information that can be used by the architect-engineer in the development of facility costs. The cost data will be used in the preparation of a cost-benefit analysis.

Atlantic Richfield Hanford Company will be involved in the GEIS for several more months in the performance of miscellaneous tasks such as:

- o Supplemental information on facility designs for development of costs.
- o Revision of the design descriptions to suit the final document.
- o Review and comment on sections prepared by others.

Nuclear Regulatory Commission S-3 Task Force On
Waste Management Impact for Reactor Licensing -
J. R. LaRiviere

The Office of Nuclear Material Safety and Safeguards, U. S. Nuclear Regulatory Commission, issued the final report on the S-3 Task Force on waste management impact for reactor licensing (NUREG-0116, SUPP 1 to WASH-1248, "Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle," a Task Force Report, NRC, dated October 1976).

Backup calculations and data in support of the S-3 Task Force work performed by ARHCO personnel were transmitted to NRC in October. (Letter, October 1, 1976, J. R. LaRiviere to W. P. Bishop).

2.0 SEPARATIONS, PRODUCTION AND SCRAP RECOVERY

2.1 SEPARATIONS - PUREX

The objectives of this program are to provide technical support for reprocessing N-Reactor Fuel at Purex to recover uranium, plutonium and neptunium while adhering to safety and environmental release requirements. Updating of the technical manual, automatic control systems for the vacuum fractionator and process condensate recycle system, and improved operating methods for acid absorbers are to be completed prior to plant startup (September 1978). Methods for controlling NO_x and ^{129}I emissions and improving waste denitration methods will be developed for possible installation during plant operation.

Flowsheet Development - D. W. Jeppson

The Purex flowsheet for reprocessing N-Reactor fuel was updated to include process modifications since shutdown, changes in fuel to be processed, and release of radioactive isotopes to the environment. The flowsheet ARH-F-103, "Purex Flowsheet - Reprocessing N-Reactor Fuels", was issued and this project is completed.

Technical Manual Revision - D. W. Jeppson, G. J. Raab

The Purex Technical Manual is being revised to include reprocessing of N-Reactor fuel and to incorporate technical bases for plant changes which have been adopted into the plant since initial startup. Assignments of sections have been made to individual contributors.

Improved Vacuum Fractionator Control - R. K. Owen

Work has begun to identify the process variables most amenable to automatic control which will provide a more consistent (50 ± 1 percent HNO_3) bottoms product while maintaining less than $.001\text{M}$ HNO_3 in the overheads. It is expected that the instrumentation for control will be selected and ordered during the next reporting period for cold pilot plant testing.

Acid Mixing for HA Column Scrub - C. C. Meinhardt

A vendor survey is being conducted to find a conductivity instrument and automatic controller for maintaining constant nitric acid concentration (2 molar) in the HA Column scrub. Promising instrumentation will be purchased and cold tested in the pilot plant.

 NO_x Control - S. S. Koegler

An appropriation request has been approved to purchase a monitor capable of indicating NO and NO_2 concentrations. The monitor will be used in conjunction with a pilot scale bubble cap column to evaluate conditions which might improve PUREX NO_x emissions. Most column components for the pilot plant column including rotameters, heat exchangers, and bubble cap trays, are now on order. A computer model of the existing two-bubble-cap tower system is being written. The program will be useful for optimizing Purex and pilot plant tests.

Iodine-129 in Purex - D. M. Strachan

Purex silver reactors have been analyzed in an effort to determine if ^{129}I was controlled by removal as silver iodide. In the last quarter it was established that these reactors did not retain ^{129}I . More detailed analyses of the Purex

silver reactor contents (berl saddles) indicated the presence of a volatile antimony-125 (^{125}Sb) species in the dissolver off-gas. This was determined by analyzing the berl saddles for as many isotopes as possible and for the amount of uranium. If the isotopes including ^{125}Sb got to the silver reactor by entrainment, then the ratio of the isotope activity to uranium should be similar to that calculated in irradiated and aged fuel. To serve as a basis for comparison, ratios were calculated using five-year-aged fuel of 0.95 percent enrichment and irradiated to 2200 MWd/t. While the actual average fuel that has been dissolved is somewhat less irradiated than 2200 MWd/t, the comparison used here remains valid. Table I illustrates the difference between the $^{125}\text{Sb}/\text{U}$ ratio and the ratios for other observed isotopes.

In all cases the $^{125}\text{Sb}/\text{U}$ ratio is higher for the saddles than calculated from the irradiated and aged fuel, while the other isotope/uranium ratios are lower in the saddles than in the fuel. Lowering the average fuel burnup for the calculated ratios improves the agreement for all isotopes except antimony which gets worse.

Experiments are under way to determine if antimony is volatilized to an appreciable extent during the dissolution of zirconium and/or uranium or if the chemistry is unique to irradiated fuels.

2.2 PRODUCTION - Z PLANT

This program will provide for improved throughput capacity, alternate products, improved operating efficiency, and will minimize the salt waste produced from recovery, and meet customer specifications for product quality and product forms.

TABLE IISOTOPE TO URANIUM RATIO
PUREX UPPER "B" SILVER REACTOR

Isotope	Ratio (k Bq/mgU)	
	Berl Saddle	Aged Fuel
^{125}Sb	96	9
^{90}Sr	58	230
^{106}Ru	20	46
^{134}Cs	3	20
^{137}Cs	184	270
^{144}Cs	12	85
^{154}Eu	0.8	1.4
^{155}Eu	2	2

Americium Recovery Flowsheet - L. A. Fort

The purpose of this work is to develop a flowsheet for the Americium Recovery portion of the Waste Treatment Facility.

Preliminary design and scope for the replacement facility for the ion exchange americium recovery system in the 242-Z Building is being pursued. Development of a flowsheet for the replacement facility is continuing as information on the types of replacement equipment to be installed is defined.

Solvent Extraction Model - L. A. Fort

The purpose of this activity is to develop a mathematical model of the Plutonium Reclamation Facility (PRF) solvent extraction system using the SEPHIS computer program.

Previous computer results for evaluating trichlorobenzene (TCB) and tetrachloroethylene (TCE) as possible replacement diluents for carbon tetrachloride (CCl_4) showed that under conditions identical to the standard plutonium(IV) flowsheet, extraction column losses would double for TCB but decrease by a factor of 1000 for TCE. The stripping column losses would decrease by a factor of 10 for TCB and increase by a factor of 200 for TCE using a four theoretical stage model of the column.

During the quarter, the SEPHIS program was again altered for evaluating a mixture diluent of 30 percent TCE and 70 percent TCB. Results show that the mixture diluent behaves very closely to CCl_4 .

A review of the number of theoretical stages actually present in the stripping column presented some enlightenment on the behaviors of the various alternate diluents studied. Experimental findings showed that six theoretical stages

may be more representative of the stripping column than four. With this information, the waste losses seen for the stripping column are caused by the degradation products of the extractant and the diluent has effect only on the column profile. With this information and the modeled profile, the inventories of the solvent extraction columns were calculated for the studied diluents and listed in Table II for CCl_4 , TCB, and TCE.

TABLE II

CALCULATED COLUMN PROFILES AS A
RESULT OF ALTERNATE DILUENTS

	<u>CCl_4</u>	<u>TCB</u>	<u>TCE</u>
Extraction (Kgs)	1.6	1.8	1.1
Stripping (Kgs)	1.7	1.5	2.2

The results show that under standard conditions, tetrachloroethylene may be better suited as an alternate than trichlorobenzene since the stripping column waste losses are an effect of only the solvent degradation products.

Plans call for further evaluation of the diluents at different flowrates and stream compositions. The program milestone date of first quarter fiscal year 1977 for selecting a new diluent based on extraction characteristics is on schedule. Work will be delayed on modeling and documenting the standard plutonium-uranium partitioning flowsheet.

Waste Treatment Columns - L. A. Fort, C. E. White

The purpose of this work is to develop improvements in the Waste Treatment Facility solvent extraction equipment and flowsheet for improved recovery of americium and plutonium.

Analyses of the Waste Treatment Facility extraction columns have shown that more americium can be recovered by reducing the acid concentration of the feed and the aqueous-to-organic flow ratio. The flow ratio change would have greater significance in the improved recovery of americium than the proposed new extraction column systems.

Pilot column waste losses and concentration profiles were determined for different operation conditions. Laboratory analysis of cerium(III) [substitute for americium(III)] was improved, permitting easier evaluation of test data.

During this quarter, a cerium(III) concentration gradient was discovered in the pilot column feed tank. Agitation of the tank during test runs along with a larger feed bath size is expected to improve column equilibrium during future test runs.

Plans call for varying flowrates, pulsing conditions, and column packing. Testing of a proposed bidentate flowsheet is also planned.

2.3 SCRAP RECOVERY - Z PLANT

Plutonium Scrap Dissolution - C. E. White

The purpose of this work is to develop optimum dissolution and/or leaching conditions compatible with Plutonium Reclamation Facility operations for recovery of plutonium from scrap.

The splashing action at the top of the prototype plutonium acid dissolver was reduced and a solids cleanout port was installed.

During the quarter, testing of the dissolver using inerts to simulate plutonium dioxide was completed. Preparations to relocate the dissolver in a glove box were made, and permission to use plutonium oxide during testing was requested.

Plans call for testing the prototype dissolver using various forms of plutonium dioxide (button oxide, oxalate oxide, and ash with various sintering temperatures). Dissolution rates and product solution plutonium concentrations are to be determined. The program will fall behind schedule because of Z Plant activities.

Bidentate Extraction - W. W. Schulz, D. C. Bouse

The purpose of this research activity is to develop a bidentate organophosphorous reagent (such as DHDECMP- dihexyl-N, N-diethylcarbamyl methylene phosphonate) for coextraction of americium and plutonium from the extraction raffinate of the Plutonium Reclamation Facility.

Resin Purification. Purification tests of technical grade DHDECMP extractant continued. The purification procedure of washing the 30 percent DHDECMP-TCB solvent once with an equal volume of ethylene glycol at 25° C, followed by passage of the washed solvent through a bed of OH-form Amberlyst A-26 resin was successfully demonstrated. Increase of resin bed temperature to 60° C did not improve its performance. Limited additional tests show that the purification scheme will satisfactorily purify DHDECMP in TCE.

Mixer-Settler Flowsheet Tests. Mixer-settler flowsheet tests with 30 percent DHDECMP-TCB solvent have continued. Using four mixer-settler stages and an aqueous-to organic flow ratio of 1.0 : 1.0 showed that 95+ percent of the americium is extracted and essentially all of the soluble plutonium in the feed is extracted. With an aqueous-to organic flow ratio of 3.5 : 1.0, 75 percent of the americium and 85 percent of the plutonium is extracted. The current Waste Treatment Facility is operated at a flow ratio of 3.5 : 1.0.

Future work will emphasize mixer-settler runs using 30 percent DHDECMP-TCB and 30 percent DHDECMP-TCE solvents purified by Amberlyst A-26 resin. Supplementing these runs the SOLVEX computer program will be used to test various flowsheet conditions. The program milestone date of first quarter FY 1978 is on schedule for completing development of the bidentate flowsheet.

3.0 CURRENT WASTE MANAGEMENT (DEFENSE)

3.1 WASTE FRACTIONIZATION

The waste fractionization program is designed to improve strontium recovery, cesium recovery, and purification process performance. Increased process efficiency will reduce strontium and cesium waste losses and cut processing costs at B Plant.

Strontium Recovery - J. W. Gehrke, J. S. Buckingham

Solvent Extraction of Sulfate Rework. A plant process test program to reduce solvent extraction strontium losses on sulfate containing rework feed is complete. Sulfate purification rework feed containing high sulfate concentrations has been processed meeting the goal of less than five percent strontium losses. A final evaluation should be written next quarter; no further work is planned.

Current Acid Waste Flowsheet. The Current Acid Waste (CAW) Strontium Recovery flowsheet will be combined with CAW cesium recovery. The proposed cesium recovery process will be incorporated into the "Tentative B Plant CAW Strontium and Cesium Recovery Flowsheet" to be issued next quarter. A final flowsheet, which may include changes resulting from pilot plant testing, should be completed on schedule, second quarter FY 1978.

First Sulfate Strike. In an effort to decrease strontium waste losses from the first strike of the sulfate purification process, a lead carrier precipitation process was developed. The laboratory test recovered 90 percent of the

strontium from the first strike waste. A plant process test has been prepared and is expected to be tested during the next three months.

Metathesis Step. The second phase of this program is to reduce the strontium waste losses from the metathesis step of the sulfate purification process. Laboratory tests have been completed that indicate none of the metathesis process variables may have a significant effect on the centrifuge efficiency and the resulting losses of strontium-bearing solids. An additional four to six months of laboratory study would be required to more precisely measure the effect of process variables on strontium losses. No further laboratory work is planned.

Cesium Recovery - R. F. Carlstrom, J. S. Buckingham

The objective is to provide an accurate description of Cesium Ion Exchange recovery through a flowsheet document, test various ion exchange resins, and develop an ion exchange computer program.

Ion Exchange Removal of Cesium from Current Acid Waste (CAW). Laboratory work showed that cesium can be removed from CAW absorbing the cesium on Zeolon-900, a product of Norton Company, after removing strontium by solvent extraction. Maximum cesium absorption occurred between pH 8.5 and pH 4. Laboratory ion exchange column runs using synthetic B Plant CAW showed a five percent cesium breakthrough occurred at 92 column volumes. Decontamination factors of 4800 and 1200 were found for aluminum and iron, respectively.

Ion Exchange Flowsheet. The Ion Exchange Flowsheet has been reviewed and edited, and the final flowsheet has been submitted for approval and publication. This flowsheet which covers the Purex Sludge Supernate (PSS) cesium recovery process in B Plant will be helpful in development of the final CAW flowsheet.

Ion Exchange Resins for Removal of Cesium from Current Acid Waste. The Chemical Technology Laboratory has completed its investigation of various zeolite resins for use in cesium removal from CAW. The resin Zeolon-900 has been recommended for use in the cesium recovery system at B Plant. Based on this recommendation, Development Engineering is organizing the pilot plant project to test the resin and aid the development of the CAW flowsheet. The design of the pilot plant system has been completed and procurement of necessary equipment is in the final stages.

B Plant Computer Programs - J. W. Gehrke

Solvent Extraction Computer Model. The solvent extraction computer model provides an accurate and convenient means for flowsheet modification and an engineer training tool. No further development work on the computer model is planned at this time. Applications of the program have provided a reliable prediction of major flowsheet changes.

Ion Exchange Computer Program. A computer model and program for ion exchange processes is being developed. The computer program is specifically aimed at Zeolon-900 resin systems, particularly the proposed CAW cesium recovery flowsheet. Pilot plant data will be used to update the program, as it becomes available, to fit the CAW process. Testing of

the program with cesium purification process data has been favorable. The present operating system in B Plant utilizes a small Zeolon-900 column.

Computer Programs User's Manual. A user's manual will be prepared which will describe the use of the solvent extraction, ion exchange, and other B Plant computer programs. The manual format will allow revisions and updating as new process control programs are developed. The document should be completed third quarter FY 1977.

3.2 ENCAPSULATION

The purpose of this program is to provide the process technology necessary to solve equipment and process problems existing in the Waste Encapsulation and Storage Facility (WESF). New equipment will be developed to replace unreliable plant equipment so encapsulation production goals can be met.

Tilt-Pour and Pan-Dry Cesium Equipment - J. D. Moore

Development of the tilt-pour melt caster equipment for encapsulation of cesium chloride continued with demonstration of operations. In ten nonradioactive tests conducted with hot cell manipulators, successful demonstrations of solution boiloff, salt melting, casting, and capsule removal were made with the tilt-pour equipment. A new feed tank was fabricated and design modifications to the melter capsule holder and support stand were completed and tested.

Design of a modified stirring system for the pan dryer was completed, but further activities with this alternate equipment for cesium chloride solidification have been

postponed due to budgetary constraints. In the next quarter, development of the tilt-pour equipment will be extended to include remote reliability and maintenance tests.

Alternate Compounds for Encapsulation - D. M. Strachan

Compounds which have chemical and physical properties and which are more compatible with future waste management plans are being sought. An alternative to SrF_2 is Sr_2SiO_4 , which has the same strontium density as the fluoride. This compound has been prepared using a strontium nitrate solution and either an ammonia stabilized silica sol (Ludox AS-40 DuPont de Nemours and Co., Wilmington, Delaware) or silicic acid. This mixture was dried, calcined at 700°C to decompose the nitrate, and fired at 1000°C to yield Sr_2SiO_4 .

In question was the water solubility of this compound. During this report period the solubility was determined to be about 1 gram per liter. While this is substantially higher than the solubility of the fluoride, a process is proposed which is simpler than that used for the fluoride and which avoids a precipitation step. Assuming a higher strontium-to-sodium ratio than is presently fed to the encapsulation facility, the incoming feed would be concentrated to 2.9M in strontium. Without cooling, a stoichiometric amount of silica would be added and the mixture evaporated to dryness. This would then be calcined and fired, or if hot pressing equipment is in place, calcined and hot pressed. This material may hot press to theoretical density easier than SrF_2 .

In an effort to improve the leach rate of the strontium alternate compound, a process similar to the Clay Calcination process is being investigated. In this process, incoming feed from B Plant would be mixed with clay (kaolinite, bento-

nite, etc.), calcined to destroy the nitrates, and fired. Preliminary work indicates that hard, dense products can be made with strontium contents greater than 60 mass percent and leach rates of about 1×10^{-5} g strontium per square centimeter surface area per day (about the same as glass). These products will be better defined over the next quarter.

Pollucite ($\text{CsAlSi}_2\text{O}_6$) has been established as the best alternative to cesium chloride for encapsulation. In recent work the feasibility of using high montmorillinite clays in the reaction has been studied. Although not all of the analytical data have been gathered, preliminary indications are that the reaction between CsCl or Cs_2CO_3 and these clays yield products which contain greater than 30 to 35 mass percent cesium and have leach rates based on cesium which are in the low 10^{-8} g $\text{Cs}/\text{cm}^2/\text{day}$ range. In fact, some have leach rates which are lower than the limits of detection.

Capsule Decontamination by Electropolishing - D. W. Jeppson

Capsule decontamination by electropolishing has been developed to reduce the in-cell processing time required to decontaminate capsules of strontium fluoride and cesium chloride. Improvements have been made on the anode (capsule holder) to provide a longer life and minimize time spent replacing electrical contacts between the capsule and the capsule holder. Corrosion rates have been measured on 304L and 316L in phosphoric acid containing chloride and fluoride to evaluate the life of the electropolishing tank and in-cell piping. A hazards review committee has been formed and is making a safety review. The final development report and safety analysis reports are being prepared.

Hot Pressing Strontium Fluoride - D. W. Jeppson

An alternative process for encapsulating strontium fluoride is being developed. Two additional test pressings have been made and evaluated. The highest density obtained using non-radioactive strontium fluoride with typical WESF impurities was 97 percent of theoretical by heating a specimen to 1100° C for eight hours and then applying 2300 psi pressure. Some of the strontium fluoride extruded into the pores of the graphite mold for several millimeters.

Further test pressings are being done to evaluate the possibility of reuse of molds, application of the use of a liner and to determine the effect of a temperature gradient in the pressing which would be present when pressing radioactive strontium.

3.3 WASTE CONCENTRATION

The objectives of this program are to transfer high-level waste into less mobile solids, and to safely isolate these solids from the biosphere for an extended time.

Waste Concentration Process Technology

Heat Transfer from Solidified Waste Tanks - J. D. Kaser

The objective of this activity is to establish criteria that will prevent tank failure from overheating due to radiogenic heat.

Steady-state temperatures were calculated for tanks containing solidified salt cake and sludge for varying thicknesses and heat generation rates for the sludge. The results have been reduced to a chart and algorithm for determining the safe amount of sludge that may be left in a tank when it is filled with salt cake.

Steady-state heat transfer calculations yield lower allowable heat generation limits than are obtained when the finite half-lives of the radioisotopes were considered by making unsteady-state calculations. However, the unsteady calculations are lengthier and more expensive. One transient calculation was made for a typical tank yielding peak temperatures that are 85 to 90 percent of the temperatures calculated from a corresponding steady-state model.

The calculational error caused by the approximate numerical method used by the computer code (HEATING 4) is being checked by comparing transient and steady-state temperatures obtained by the computer code with analytical solutions using a finite heat generating cylinder as a test case.

Waste Management Model Studies - W. C. Schmidt

The objectives of these studies are to evaluate various processing alternatives and to identify future waste volumes which provide a basis for future needed facilities.

Six alternative modes of processing residual liquor with the partial neutralization and double-shell slurry processes were evaluated with the Waste Management Model. The volumes of stored high-level wastes were projected through FY 1985 for each of the alternatives. A topical report describing these studies is in preparation.

Interstitial Liquor Removal - W. E. Ogren

The objectives of this activity are to develop and demonstrate methods for removing as much interstitial liquid from Hanford waste storage tanks as is technically and economically practicable and to immobilize the residual liquid.

Theoretical and experimental studies into the feasibility of the capillary pump concept continued. The studies will provide a predictive tool for the rate of imbibition of interstitial liquors into high capillary potential materials. Calculations indicate diatomaceous earth has a capillary potential which may remove drainable liquors from salt while contaminating only four weight percent of the amount of silica which would be contaminated by leakage into Hanford soil. Documentation of the imbibition studies is planned.

Laboratory and engineering studies of Ostwald ripening of the Hanford salt cake crystals are underway. With time, large soluble salt crystals grow at the expense of small salt crystals, changing the capillary characteristics of salt cake.

Evaluation of *in-situ* drying is underway with an engineering study of rates, and material and energy balances. Process testing of *in-situ* drying is planned based upon results of the engineering study.

The 241-TK-111-S jet pump salt well has been in operation periodically. An ultrasonic flowmeter and a solid-state pressure transducer have been installed for evaluation purposes on the salt well. Production of interstitial liquor from this well during this quarter has totalled 27,000 gallons at an average pumping rate of two gallons per minute.

A flowsheet concept for application of alternative methods of interstitial liquor removal/stabilization has been defined. Choice of the appropriate interstitial liquor removal/stabilization technique is based upon stored waste properties as perceived by photographs, samples, psychrometric data,

jet pump salt well production performance, and liquid levels in observation wells. Additional engineering study of the flowsheet concept is planned.

Residual Liquor Disposition - R. E. Van der Cook

The objectives of this activity are to develop and demonstrate processes for converting residual liquors to acceptable solids for interim storage, preferably in existing underground waste storage tanks.

The Double-shell Slurry Process Test at 242-S Evaporator-Crystallizer was delayed from the scheduled late November 1976 date since the double-shell tanks required for the process test were not available due to the work stoppage. This delay in the process test means that the six double-shell tanks provided for in the FY 1977 budget will be built.

A second pilot plant double-shell slurry test of liquors simulating those expected from partial neutralization indicates that double-shell slurry is a workable process. Greater volume reductions, fewer solids, and a lower boiling point rise were found than those in the previous batch laboratory tests.

Construction work on the partial neutralization prototype at 242-S Evaporator-Crystallizer resumed. The current estimated date for completion is in late February 1977. Partial neutralization flowsheet documentation proceeded with the issuance of a working draft pending final editing. A draft report of the hazards review team findings has been issued. The major impact is the need to reroute the pump seal water drains from the pump room sump directly to the sump drain line. This rerouting will assist in detect-

ing an acid leak into the sump and will allow additional time for neutralizing the acid and subsequent disposition before the evaporator must be shut down.

Additional laboratory studies to characterize the stoichiometry of nitrogen dioxide (NO_2) conversion to nitric oxide (NO) upon dilution of concentrated nitric acid were undertaken. The purpose of these studies was to further quantify conversion trends apparent in the initial scouting runs. Equipment problems with both of the available NO_x monitors precluded gaining significant results. Early resolution of the instrument problem is required since one of the NO_x monitors is intended for plant use.

Laboratory results from the evaporation of actual and synthetic ITS 1 and 2 and 242-T bottoms liquors were compared to the results forecasted by the computer program PREDICT. The volume reduction data from laboratory boildown tests and from the computer program PREDICT agree within 15 percent with two exceptions out of six values. The volume percent solids formed agree within 10 percent with two exceptions out of six values. Thus, the PREDICT program can be considered as a reasonably reliable tool for forecasting the characteristics of evaporator bottoms product. Details are presented in a recent engineering study.

Salt Cake and Sludge Characterization - G. K. Allen

The objective of this subtask is to obtain core samples of waste tank solids for the purpose of defining chemical and physical properties of the solids stored in waste tanks as a function of depth or strata.

The salt cake core driller has drilled through approximately 20 feet of solids in 241-TX-116 as part of the testing of the drilling equipment in a radioactive environment. The percent volume of core that has been recovered to determine chemical and physical properties of the solids has been low; however, adequate samples were obtained for laboratory analysis. The drill string is scheduled to be removed from the tank for drill bit examination and evaluation. After the drill string is removed, further cold testing will be conducted to determine the causes of the low core recovery.

3.4 TANK FARM SURVEILLANCE AND MAINTENANCE - R. G. Oliver

The objective of this program is to develop improved instrumentation and surveillance methods for waste tanks and transfer lines.

A contract was signed in April 1976 with the Boeing Company for the development of a soil conductivity prototype leak detection system. This system was to be installed on five tanks in the "A" Tank Farm complex to demonstrate that the theory of operation developed under a previous research contract could be put into practice in an operating tank farm environment. Because of onsite labor negotiations, considerable program slippage occurred and the contract was extended through December 23, 1976. Initial tests of the second generation design conducted on Tank 241-AX-102 were inconclusive. The development program was modified to investigate the problems encountered and significant improvements to the system were made. New sensitivity tests have been successfully completed with excellent preliminary results.

Of immediate concern is the lack of budgeted funding for project completion. The contract extension will complete the evaluation of the sensitivity testing just concluded.

The first demonstration "in-place" gamma monitoring system is complete. Initial laboratory and field operating tests have been very successful. The system is comprised of remotely monitored sensitive GM type gamma detectors located at intervals within a dry well casing in such a way that use of the dry well for scanning by conventional equipment can continue as required. The in-place detectors are sequentially scanned by a computer controlled counting system. The computer performs a statistical analysis of the data received and provides an alarm where a problem or a trend exists. The demonstration system can scan 45 detectors located up to two-thirds of a mile away. It was originally scheduled for a test installation in nine selected dry wells surrounding Tank 241-AX-108. Due to lack of onsite contractor craftsmen, this installation was cancelled. A hand-installed entry to each dry well was made to verify calibration sensitivity in the well environment. The results show that the system is capable of detecting very minor radiation level changes in the wells. After compensation for natural radiation level variations that normally occur, detection of an increase of about 1 to 2 $\mu\text{r/hr}$ at a nominal background of 8 $\mu\text{r/hr}$ can be realized. This sensitivity level appears adequate to detect early ^{106}Ru fronts with the detector spacing provided. An extended test installation of the system will be made in wells around Tank 241-A-102 and the existing computer will be replaced with a micro-computer. It should be noted that this system appears equally suited for any other normally static surveillance activity requiring great sensitivity and a large number of dispersed detectors.

Development work is still in progress on a solid-state electronics pressure transducer and readout system that has shown satisfactory resistance to "salt-over" while measuring interstitial salt well liquid levels. The prototype system has operated successfully in simulated salt cake and sludge environments. Tests in the salt well system of Tank 241-S-111 have been in progress since November 18, 1976. Readings do not agree with the dip tubes but do agree with neutron probe determination of the liquid level. The dip tubes are suspected of being plugged or faulty. This is to be confirmed by a direct liquid measurement using a conductivity probe.

A portable flowmeter using an ultrasonic measuring technique has been successfully tested under a variety of conditions. It is currently installed on the salt well pump of 241-S-111. Pumping started December 9, 1976. Agreement with the magnetic flowmeter was better than one percent at approximately 10,000 gallons of transferred liquid. Additional testing showed unacceptable discrepancies that have been traced to the ultrasonic flowmeter. The problem is being corrected, and testing will continue. The portable system is useful for temporary across-the-ground transfers and for salt well pumping flow measurements. A computer-based version of this system utilizing up-down totalizers is recommended for permanent installation on selected transfer lines should the tests be successfully concluded.

Development of a neutron soil-moisture monitoring detector for scanning surveillance dry wells is now about 80 percent complete. Assembly can be completed as soon as the parts are fabricated at the shops. The detector is to be used with the dry well surveillance vans. It is hoped that the detector will be sufficiently free of gamma response to also allow its use in the internal dry wells of tanks containing

salt cake or sludge. Interstitial liquid levels and dry salt cake moisture content measurements may be possible.

A high/low gamma detector probe has been successfully tested in the surveillance leak detection pit of Tank 241-AZ-102. The probe contains two GM type detectors operating together to scan for very low levels of radiation but remaining operational if very high levels are encountered. Normally sensitive gamma probes would be overwhelmed immediately and cease to function in the event of significant leakage. This probe can also be used in conjunction with the "in-place" computerized gamma detection system and is especially useful for surveillance of leak detection pits, diversion boxes, pump pits, lift stations, and the like. This development was completed December 1, 1976, with the installation of the final configuration on Tank 241-AX-102. Other units are being prepared for installation in the leak detection pits of the remaining tanks in 241-AX Farm and also in 241-AY and 241-AZ.

4.0 LONG-TERM MANAGEMENT OF DEFENSE WASTE AT HANFORD

The overall objective of this mission is to develop, evaluate, and demonstrate the technology for the long-term management of the Hanford high-level waste which will remain upon completion of the current waste solidification program. This waste will consist of approximately 3,500 capsules of ^{90}Sr and ^{137}Cs stored in water basins and 6,000,000 cubic feet of material in underground tanks. Because the waste will remain hazardous for hundreds to thousands of years, a long-term waste management program is needed to ensure its long-term isolation from man's environment. Development and demonstration of the technology required to implement this program is expected to be complete by the mid-1980s.

The alternatives being considered for Hanford high-level defense waste are:

- To retrieve the waste, process it to an improved stabilized form, and store it retrievably in deep natural rock or other formations (geologic storage).
- To retrieve the waste, process it to an improved stabilized form, and store it in new retrievable surface storage facilities for an interim period, with possible eventual transfer to a final storage site.
- To continue storage of the waste in existing tanks and capsules as is, or with improvements to the storage facilities to extend the period of safe storage.

The mission has been divided into five major tasks:

- Storage System Integrity and Engineered Improvements,
- Waste Retrieval,
- Waste Processing,
- Storage and Disposal Alternatives, and
- Alternatives Studies and Assessments.

4.1 STORAGE SYSTEM INTEGRITY AND ENGINEERED IMPROVEMENTS - P. F. Mercier

The objectives of this program are:

- To develop a data base to permit the annual assessment of the safety of continued storage of solidified waste in the existing single-shell tanks; to be completed in FY 1980.
- To identify, develop, and test engineered improvements that would extend the safe storage period of the single-shell tanks; to be completed in FY 1980.
- To analyze the safety and the effect on the biosphere of continued solidified waste storage in the single-shell tanks; to be completed in FY 1980.

Although the presently solidified waste is immobile compared to the original liquid form, there is concern about the continued reliability of the tanks because of the potential hazard to the biosphere in the event of loss of containment. The first annual assessment of the safety of continued waste storage in the Hanford tanks is planned for publication at the beginning of FY 1979. Testing and selection of engineered improvements to the storage tanks are scheduled for completion in FY 1980.

Storage System Integrity

The following activities are planned to provide a data base to assess the integrity of the single-shell waste storage tanks and to develop methods to periodically evaluate their integrity.

Salt Cake Stability - G. A. Beitel

The nitrates which form the bulk of salt cake are energetically capable of reacting with any organic materials which may be present. Safe storage of high-level wastes such as salt cake requires that violent or uncontrolled exothermic reactions between the salt cake and any organic materials present be kinetically impossible.

Tests have shown no evidence of susceptibility to shock-initiated detonation or to combustion within the moisture - nitrate - organic mixture calculated to be present.

A proposal has been received from Hazards Research Corporation to conduct tests to determine the reactivity of salt cake under conditions of programmed temperature increases and under conditions of extended incubation at constant temperatures.

Three types of reactions can lead to explosions: detonations, condensed phase deflagrations, or uniform exothermic reactions leading to thermal explosion. In terms of damage potential, detonations generally are more devastating than deflagrations or thermal explosions. However, the compositions which can detonate are generally more restricted than those which can deflagrate or undergo thermal explosion. Thus, all compositions that can detonate can deflagrate; whereas, some compositions that can deflagrate may not detonate. Likewise, all compositions that deflagrate can

undergo thermal explosions; while some compositions that can react to produce thermal explosion may not deflagrate. Since any kind of explosion would be equally unacceptable in these circumstances, the most important aspect is to demonstrate the susceptibility of the cake to undergo an accelerating exothermic reaction which could lead to a thermal explosion. Therefore, the approach selected for this study is to determine the reactivity of the salt cake under conditions of programmed temperature increases and under conditions of extended incubation at constant temperature.

The temperature at which a thermal explosion is initiated is determined in a 350-ml, heavy-walled, stainless steel reactor capable of withstanding the maximum pressure that can result from the decomposition reaction. The reactor is equipped with a thermocouple and a rapid response pressure transducer.

Initially, a screening experiment is performed in which the sample is heated at a rate of approximately 1°C per minute until reaction occurs. When the sample reacts, the maximum pressure generated and the rate of pressure rise are obtained.

A fresh sample is then incubated for 24 hours at a temperature 10°C below that at which decomposition occurred. If decomposition occurs again within 24 hours, the operation is repeated until the maximum temperature at which decomposition does not occur after a 24-hour incubation period is established.

The sample size is ordinarily 20 grams. Depending on the anticipated severity of the decomposition reaction, larger or smaller samples may be employed.

The samples to be studied will include compositions that simulate the estimated composition of the undrained and drained salt cake, including organic complexant chemicals and ion exchange resin. The nominal composition for the undrained and drained cases is given in Table III. In addition, the concentration of the organic component will be varied up and down by a factor of 2 in order to determine the effect of concentration gradients which may have developed during the storage and drying out period. Three compositions of the ion exchange resin with sodium nitrate salt cake will also be studied. Thus, the overall scope of the program includes the study of three concentrations of the organic complexant in the undrained salt cake, three in the drained salt cake, and three concentrations of ion exchange resin with the salt cake, for a total of nine separate compositions.

Contract procedures are in progress, and testing is expected to begin in early January. A final report will follow within five months after start of testing.

Containment Integrity - F. R. Vollert

The objective of this activity is to determine the integrity of the reinforced concrete tanks. The extended use of the waste storage tanks will depend on how seriously elevated temperature, thermal cycling, and chemical solutions have affected the elastic properties of the concrete and the reinforcing steel.

Two laboratory programs have been developed by the Portland Cement Association Laboratories (PCAL) to investigate the effects of elevated temperatures and simulated waste on specimens of the kind of concrete used in Hanford tank construction. All specimens were manufactured with Hanford aggregate and the original 3,000 psi and 4,500 psi concrete formulations.

TABLE IIIESTIMATED COMPOSITION OF SALT CAKES
CONTAINING ORGANIC COMPLEXANTS

Basis: 1 liter of cake

COMPONENT	UNDRAINED ⁽¹⁾	DRAINED ⁽²⁾
	(FULLY SATURATED) GRAMS	(RESIDUAL SATURATION) GRAMS
NaNO ₃	1,250	1,150
NaNO ₂	100	40
NaAlO ₂	100	40
NaOH	60	24
Complexant		
Carbon	150	60
Hydrogen	20	8
Oxygen	140	56
Nitrogen	70	28
Sodium	100	40
Water	250	100
Total Weight	2,240	1,546

(¹) Assumes 50 volume percent solids, 50 volume percent liquor containing 2.5M complexant

(²) Assumes 50 volume percent solids, 30 volume percent air, 20 volume percent liquor containing 2.5M complexant

The fourth quarter Progress Report of the Thermal Effects Program shows a modest decrease in the elastic properties of the concrete. The program is continuing into its second year.

Concrete cores taken from in-service tank domes during past tank modifications have been cut into 3" x 3" cube and 3" x 6" cylinder specimens. Elastic modulus, compressive strength, Poisson's Ratio, and splitting tensile strength tests will be made at room temperature and at 250° F after 3, 30, 150, and 270 days. Petrographic examinations will be made of aggregate - cement paste relationships in the as-received condition and after heating to 250° F for 3, 30, 150, and 270 days. Testing has begun, and progress reports will be made after completion of each test period.

Engineered Improvements

The objective of this activity is to identify, develop, and demonstrate methods and techniques that can extend the safe storage period of the solidified waste in the single-shell tanks.

Tank Dome Support Studies - P. F. Mercier

The objective of these studies is to determine a feasible method to support a tank dome in the event of loss of elasticity with time or the impact of an aircraft on the earth cover.

Three concepts have been selected from 15 developed by the consultant for additional study and cost evaluation. The final report will be issued at the beginning of the next quarter.

Tank Dome Elevation Surveillance - R. G. Geier

The objective of this subactivity is to monitor changes in the elevation of the tank domes that could be indicative of cracking of the concrete or yielding of the reinforcing steel.

The program of monitoring waste tank dome elevations includes obtaining measurements at three locations on tanks 101-SY and 102-SY, at 11 locations on tanks 101-AX and 103-AX, at two locations on tanks 102-AX and 104-AX, and at one location on tanks 103-A - 106-A. Bench marks have been attached to the SY tanks, and attachment to the A and AX tanks is anticipated in January.

Initial measurements were obtained on the SY tanks using temporary markings in lieu of the permanent bench marks. Beginning in January, measurements will be taken at each location on a quarterly basis.

Tank Safety Analysis - P. G. Easley

The purpose of this activity is to determine the effects upon the biosphere of continued storage of the solidified waste in the single-shell tanks. This analysis will determine the radiologic impact upon man resulting from the loss of containment through failure of the storage system.

Work during this quarter was mostly planning. The determination of the possible transport of radionuclides from the tank farms by biota has been scoped, and the work will be done by ARHCO personnel. Two laboratory experiments were designed in cooperation with Battelle-Northwest personnel to study the movement of radionuclides in a soil column, but it was decided that BNW was overcommitted and that the work would be done offsite.

A research report was received from BNW describing the effects of temperature on the flow of water through Hanford sediments. This was of interest because it was felt that the temperature field around a waste tank would affect the flow of a tank leak. The initial results show little effect because of temperature variations from 6 to 30° C.

4.2 WASTE RETRIEVAL - I. E. Reep

The objective of this task is to develop and demonstrate the technology necessary for retrieval of the waste material (salts, sludges, liquids, and accumulated debris) from the underground storage tanks.

Conceptual design of the proposed prototype waste retrieval system will be completed in FY 1977, and fabrication of this system will begin in early FY 1979. Cold testing of major components of the waste retrieval system could take place as early as FY 1980, with hot testing and demonstration of this equipment beginning in FY 1982.

Concept and Equipment Development - I. E. Reep

Concept Development - H. A. Wallskog

This activity is nearing completion. The major components for accomplishing the waste retrieval objective have been identified. They are the (1) prototype waste retrieval system, (2) prototype test facility and, (3) facilities for hot demonstration.

The document entitled "Program of the Hanford High-Level Waste Retrieval Task - A Narrative Description", ARH-LD-144, was issued in draft form during the report period and is now awaiting clearance for offsite distribution. Issuance is anticipated in January 1977. This report describes the plan to design, construct, test and demonstrate a Prototype

Retrieval System having the capability to safely remove solidified waste from any of the large underground storage tanks.

Equipment and Tool Development - N. M. Arnold

The objective of this activity is to develop tools and equipment for recovery and removal of wastes from the underground tanks. Work is currently being focused on the development of hydraulic retrieval devices.

Two Scumbuster (Vaughn Company, Inc., Montesano, Washington) agitator pumps are being evaluated for their capability to slurry salt cake in its own interstitial liquor and pump the resulting slurry out of the tank. This technique could also be used for the retrieval of sludge if the moisture content is sufficiently high. A hydraulic retrieval test loop has been installed for use with the larger, 100-horsepower pump, which has successfully slurried and pumped synthetic salt cake containing up to 70 weight percent included solids.

Vaughn Model 150 "Scumbuster" Pump Test. The purpose of this test series is to determine the capability of a small centrifugal pump to remove solid radioactive wastes from underground storage with only limited additions of water. Completion of preliminary pump evaluation runs of the 60-horsepower "Scumbuster" style pump system is expected in early January, 1977. Tests of pump performance on synthetic salt cake are currently scheduled to begin in mid-January, 1977.

Pump-Bucket Recovery Tool. This tool is designed for digging through hard, crusty material. A controlled addition of water is used to form a slurry with the aid of a "Scumbuster" type pump that is also used to pump the slurry to a central distribution point prior to removal from a waste tank.

Detailed design of the pump-bucket recovery tool is continuing with intermittent delays due to work on higher priority items. The schedule for initiation of fabrication has been revised to March 1977.

Engineering Studies and Conceptual Design - I. E. Reep

This activity consists of:

- o Engineering studies in areas requiring further definition, and
- o Conceptual designs for the further development of concepts and equipment identified as feasible in the Concept and Equipment Development activity.

A number of engineering study reports have been received from consultants. These include an engineering study and cost estimate for a waste retrieval system; a design study of a salt cake retrieval system; a study evaluating the adaptability of standard mechanical, hydraulic, and pneumatic devices to salt cake and sludge retrieval at Hanford; and a study of the high-level waste management technology development program. This information will provide a basis for conceptual designs of a prototype waste retrieval system, a prototype test facility, and facilities for hot demonstration.

Engineering Studies - I. E. Reep, H. A. Wallskog,
J. K. Strickler

The objective of this activity is to evaluate the technical feasibility, cost effectiveness, and safety of retrieval concepts under study and development.

Some modifications to the existing tank dome structures will be necessary to accommodate the Prototype Retrieval System presently in conceptual design. Accordingly, a consultant agreement (CA-603) was initiated December 10, 1976, with Dr. V. B. Watwood, Houghton, Michigan, to analyze an existing tank structure having two 42-inch diameter penetrations added to the dome; one at the tank dome center and one at a radius of 20'-9". A final report is expected during the third quarter of FY 1977.

A work order was initiated with Battelle-Northwest Laboratories to develop a new concept for retrieval of Hanford high-level wastes using an underground mining approach. Purpose of the study is to develop the concept to the point where a venture type cost estimate can be prepared for inclusion in the HDWD. Preliminary results of the study will be available in January 1977.

Two methods for transferring retrieved bulk high-level waste from the Hanford waste tanks to an on-site waste processing facility are being studied: (1) transport of solids or slurry in a shielded container, and (2) transfer of slurry by pipeline. The purpose of the study is to develop technically feasible concepts and cost estimates and to evaluate potential safety hazards associated with transportation of high-level bulk waste.

Conceptual Design - H. A. Wallskog, N. M. Arnold

The objective of this activity is to provide conceptual designs of the major components and support facilities for accomplishing the waste retrieval objective.

Work on the conceptual design of the Prototype Retrieval System is continuing at the Programmed and Remote Systems Corp., St. Paul, Minnesota (Contract CA-190, Supplement No. 1). Further engineering studies and system design considerations have resulted in a "T" shaped platform. This mobile platform configuration, with three tracked transporters, is planned around the use of the uniform (for all tanks) penetration locations being studied above (see Engineering Studies). The overall height of an early design concept of the support tower for the articulated arm was 94-feet to provide containment of the arm in its fully retracted position. A benefit from additional design studies has been the reduction of the mechanical arm tower to approximately 65-feet.

Conceptual design and preliminary cost estimate of modifications to the 277-T Building, 200 West Area, for use as a waste retrieval tool development facility was completed by Vitro Engineering. The facility is being considered for use in evaluating the performance of tools for the recovery and removal of high-solids content slurries. However, alternative buildings are being investigated in order to conserve program funds.

4.3 WASTE PROCESSING - D. A. Turner

The objectives of this program are to:

- Develop and demonstrate technology for converting Hanford high-level waste to a immobilized form satisfactory for long-term or final storage.
- Develop and demonstrate technology for partitioning Hanford high-level waste into a decontaminated chemical waste fraction and a low volume, high activity fraction containing the radionuclides.
- Develop economic and technical criteria against which alternative immobilization concepts may be evaluated and compared.
- Integrate Waste Processing and Storage and Disposal Alternatives task efforts.

Aqueous Silicate Process - G. S. Barney, M. J. Kupfer

The conversion of fission product wastes to solid, insoluble products suitable for long-term or final storage is currently being investigated through the high temperature alternative of the Aqueous Silicate Process - Clay Calcination. In this process dry waste materials (such as salt cake) or other dehydrated wastes are reacted with clay at 600° to 800° C to form water- and nitrate-free sodium alumino-silicate powders. These powders may then be incorporated with binders or sintered to form clay calcination products.

Fixation of Actual Sludge - M. J. Kupfer

Application of the clay calcination process to fixation of washed, dried (140° C) sludge from Tank 104-S was demonstrated. The composition of the 104-S sludge is shown in Table IV. A mixture of 50 weight percent dried sludge and 50 weight percent kaolin clay was reacted at 750° C for two hours. The reacted powder was pressed (10,000 psi) and sintered at 1200° C for two hours. Only 0.047 percent of the ^{137}Cs volatilized during the 750° C heating step and 0.037 percent evolved during the 1200° C sintering step. The leach rate in deionized water (25° C) for the sintered pellet (Table V) was similar to rates found for clay-synthetic sludge specimens when based on cesium or strontium. The rate based on ^{239}Pu is lower than that observed previously for any clay calcination product.

Sintering Behavior of Clay Calcination Product - G. S. Barney

The sintering behavior of a clay calcination product was studied. Sintering was followed by measuring the linear shrinkage of cylindrical samples versus time at 1000, 1100, and 1200° C. The samples were prepared by blending the following components:

300 g kaolin
106 g synthetic Purex sludge
182 g synthetic Redox sludge
28 g $\text{Na}_3\text{PO}_4 \cdot 12\text{H}_2\text{O}$
10 g $\text{Sr}(\text{OH})_2 \cdot 8\text{H}_2\text{O}$
20 ml 3.0M CsOH

This mixture was fired at 750° C for two hours, blended with 20.4 g of additional kaolin to improve green strength, and mixed with water. The cylindrical samples were made by pressing the wet mixture at 20,000 lb in a die. The samples were then dried at 600° C before sintering.

TABLE IV
COMPOSITION OF 104-S SLUDGE

Component	Wt%
Fe	0.1
Na	8.8
Al	42.2
Si	6.9
Mn	< 0.1
H ₂ O (measured) ^a	20.0
O ₂ (calculated)	22.0
¹³⁷ Cs μ Ci/g	60
⁹⁰ Sr μ Ci/g	1900
²³⁹ Pu μ Ci/g	12.4

^aSludge heated to 850° C. Weight loss assumed to be water of hydration

TABLE V
LEACH TESTING OF CLAY CALCINATION
PRODUCT MADE FROM 104-S SLUDGE

Cum. Time, Days	Leach Rate, g/cm ² -day (25° C)		
	Based on ¹³⁷ Cs	Based on ⁹⁰ Sr	Based on ²³⁹ Pu
1	2.5 x 10 ⁻⁵	4.4 x 10 ⁻⁵	9.7 x 10 ⁻⁷
5	4.2 x 10 ⁻⁶	1.8 x 10 ⁻⁶	2.2 x 10 ⁻⁷
12	2.0 x 10 ⁻⁶	8.8 x 10 ⁻⁷	3.8 x 10 ⁻⁷
19	1.3 x 10 ⁻⁶	7.3 x 10 ⁻⁷	2.2 x 10 ⁻⁷
26	1.1 x 10 ⁻⁶	9.6 x 10 ⁻⁷	2.3 x 10 ⁻⁷
33	8.9 x 10 ⁻⁷	7.8 x 10 ⁻⁷	2.2 x 10 ⁻⁷
40	8.6 x 10 ⁻⁷	1.1 x 10 ⁻⁶	2.6 x 10 ⁻⁷

Sintering of the samples was accomplished by placing them in a furnace preheated to the desired temperature. Samples were removed at one-hour intervals, cooled, and the diameter and length measured. The results of these measurements are shown in Figure 1. Plots of density versus time are similar to the curves in Figure 1.

The curves show that initial shrinkage of the compacts is high and rapidly decreases with time. Very little shrinkage is observed after the first hours. As the temperature is increased, the shrinkage rate increases and the ultimate density appears to increase. The large increase in the shrinkage rate from 1100 to 1200° C indicates that at some intermediate temperature a process occurs (probably melting or recrystallization of a component) which greatly increases shrinkage.

A statistically designed experiment is being developed to determine the effects of sludge composition on clay calcination product characteristics (leach rate, density, abrasion resistance). The information obtained from this experiment will permit prediction of product characteristics as a function of sludge feed composition.

Silicate Melt Process - M. J. Kupfer, W. W. Schulz

The objectives of this activity are to develop and demonstrate processes for incorporation of Hanford high-level wastes into immobile silicate-based glass forms.

FIGURE 1

SHRINKAGE OF A CLAY-WASTE MIXTURE AS A FUNCTION
OF TIME AND TEMPERATURE DURING SINTERING

Leach Testing of Sludge-Glasses. Leach testing using a Soxhlet extractor technique was initiated on sludge-glasses generated from the scouting studies described in the previous report.⁽¹⁾ The products were crushed and one gram of the acetone-rinsed and dried 40 to 60 mesh (U. S. Standard Sieve Series) fraction was leached at about 99° C using the Soxhlet extractor. Weight losses of the samples were measured after 24 hours, 48 hours, and again after 72 hours. The lowest total weight percent loss found after 72 hours was 0.58 for a glass of the following composition: 30 weight percent Redox sludge - 55 weight percent SiO_2 - 15 weight percent $\text{Na}_2\text{B}_4\text{O}_7$. By comparison, Hanford basalt rock, and typical commercial soda-lime and Pyrex glasses exhibited weight percent losses of 1.5, 3.5, and 0.50, respectively. Other sludge glasses tested by this method showed weight percent losses ranging from 0.68 to 15.2. The leach durabilities of these glasses decreased with increasing $\text{Na}_2\text{B}_4\text{O}_7$ content and decreasing SiO_2 content. Durabilities also decreased when 1-5 weight percent Li_2O (used as an aid in reducing melting temperature) was present in the glasses. Additionally, weight losses were higher for the high iron-content Purex sludge-glasses than for the high aluminum-content Redox specimens.

Cesium Volatility. Only 0.02 to 0.28 percent of the cesium volatilized when typical sludge-glass charges were heated slowly from room temperature and maintained in the molten state for two hours. Less cesium evolved from the Purex sludge-glass formulations than from the corresponding Redox charges. Additions of 1-5 weight percent Li_2O or 5-10

weight percent TiO_2 to the glass formulas also suppressed cesium volatility. The amount of cesium evolved from glasses containing synthetic sludge is about a factor of 2 to 20 times lower than for glasses made from residual liquor.⁽²⁾

Fixation of Phosphate-Bearing Sludge. Large amounts of sodium phosphate (Na_3PO_4) are associated with some Hanford sludges. Synthetic Redox or Purex sludges containing up to 50 weight percent Na_3PO_4 were converted to silicate glasses using three sludge-glass formulas that have been studied extensively. Viscosities of the phosphate-bearing melts were lower than for corresponding sludge-glass melts without phosphate. Surprisingly, only slight increases in the leachability based on cesium or strontium (25°C) were noted when the glasses contained phosphate. Soxhlet leach tests have not been performed, however.

Resistivity and Viscosity of Sludge-Glass. Electrical resistivity and viscosity curves were determined at Corning Glass Works for three typical synthetic Redox sludge glasses. The results in Figure 2 clearly indicate the effect of small additions of lithia (Li_2O) on lowering the viscosity. A good viscosity range for pouring from a continuous melter is estimated to be 50 to 500 poise. This is equivalent to a furnace temperature of 1300 to $>1500^\circ \text{C}$ for formula A, 1200 to 1500°C for formula B, and 1000 to 1250°C for formula C. Lithia additions also reduce the electrical resistivities of the glasses. The resistivities for these glasses at the temperatures indicated above are about 1-2 ohm-in. The slope

TEMPERATURE-DEGREES C.

FIGURE 2

VISCOSITY CURVES FOR REDOX SLUDGE GLASS

of the resistivity curves in the range 1000-1500° C is approximately the same as for a typical commercial soda-lime glass. This is significant since the stability of power control in an electric melter depends on this slope.

Although they have not yet been determined, viscosities and resistivities for Purex sludge-glasses are expected to be lower than for the corresponding Redox glasses.

Tests with Actual Waste. Washed, dried (140° C) sludge from Tank 104-S (Table IV) was converted to a glass using two formulations that have been studied extensively for fixation of synthetic sludge. The results in Table VI indicate that both leachabilities and cesium volatilities are somewhat lower than those observed for the corresponding glasses made from synthetic waste. Although the 5 weight percent addition of Li_2O to glass B aided in reducing viscosity, it did not reduce the volatility of cesium as previously noted in synthetic sludge-glass experiments.

Silicate glass formulations found effective for fixation of synthetic residual liquor were described previously.⁽³⁾ During the present report period, actual alkaline waste liquor from Tank 110-S was successfully converted to glass. The composition of this waste liquor (Table VII) is approximately the same as that expected for residual liquor. The charge formulation shown in Table VIII was used to convert the dried (150° C) residual liquor solids to silicate glasses at 1000 and 1200° C. A small amount of a separate Na_2SO_4 phase segregated from the glass prepared at 1000° C. Phase separation was previously observed in some tests using synthetic residual liquor that contained the same concentration of Na_2SO_4 as the 110-S sample (0.05M).

TABLE VI

GLASSES FROM 104-S SLUDGE
PREPARATION - PROPERTIES

I. MELT CONDITIONS

Formulation:	<u>Melt A</u>	<u>Melt B</u>
Wt% SiO ₂	45	42.5
Wt% 104-S	30	28.5
Na ₂ B ₄ O ₇	25	23.7
Li ₂ O	-	5.0
Melt Temperature	1200° C	1000° C

II. LEACH RATES, g/cm²-day (25° C)

Cum. Time days	<u>Melt A</u>			<u>Melt B</u>		
	Based on:			Based on:		
	<u>¹³⁷Cs</u>	<u>⁹⁰Sr</u>	<u>²³⁹Pu</u>	<u>¹³⁷Cs</u>	<u>⁹⁰Sr</u>	<u>²³⁹Pu</u>
1	2.9 x 10 ⁻⁶	8.8 x 10 ⁻⁷	3.1 x 10 ⁻⁸	1.9 x 10 ⁻⁶	2.9 x 10 ⁻⁷	4.2 x 10 ⁻⁸
5	9.6 x 10 ⁻⁸	2.3 x 10 ⁻⁷	7.9 x 10 ⁻⁹	6.3 x 10 ⁻⁷	8.2 x 10 ⁻⁸	3.5 x 10 ⁻⁸
12	4.4 x 10 ⁻⁸	1.7 x 10 ⁻⁷	1.6 x 10 ⁻⁸	3.0 x 10 ⁻⁷	5.4 x 10 ⁻⁸	1.6 x 10 ⁻⁸
19	1.3 x 10 ⁻⁷	2.0 x 10 ⁻⁷	1.3 x 10 ⁻⁸	7.8 x 10 ⁻⁷	5.6 x 10 ⁻⁸	9.5 x 10 ⁻⁹
26	3.8 x 10 ⁻⁸	1.2 x 10 ⁻⁷	1.8 x 10 ⁻⁸	2.0 x 10 ⁻⁷	2.0 x 10 ⁻⁸	1.2 x 10 ⁻⁸

III. CESIUM VOLATILITY

Melt A - 0.05%

Melt B - 0.06%

TABLE VII

COMPOSITION OF 110-S RESIDUAL LIQUOR

Component	M
NaNO ₃	2.7
NaOH	3.6
NaAlO ₂	1.2
NaNO ₂	1.7
Na ₂ SO ₄	0.05
¹³⁷ Cs (μCi/l)	5.2 x 10 ⁵
⁹⁰ Sr (μCi/Ω)	1.2 x 10 ⁴

TABLE VIII

GLASSES FROM 110-S RESIDUAL LIQUOR

I. CHARGE FORMULATION

	Wt%
SiO ₂	55
110-S Solids ^a	30
B ₂ O ₃	15

II. LEACH RATES, g/cm²-day (25° C)

Cum. Time days	1000° C Melt		1200° C Melt	
	Based on:		Based on:	
	¹³⁷ Cs	⁹⁰ Sr	¹³⁷ Cs	⁹⁰ Sr
1	9.3 x 10 ⁻⁵	2.2 x 10 ⁻⁴	6.6 x 10 ⁻⁷	4.0 x 10 ⁻⁶
5	9.5 x 10 ⁻⁶	3.8 x 10 ⁻⁶	1.4 x 10 ⁻⁷	1.4 x 10 ⁻⁷
12	2.3 x 10 ⁻⁶	5.2 x 10 ⁻⁷	7.1 x 10 ⁻⁸	6.0 x 10 ⁻⁸
19	1.0 x 10 ⁻⁶	2.2 x 10 ⁻⁷	-	-

^aSolids prepared by evaporating residual liquor at 150° C

Approximately 0.6 percent of the ^{137}Cs and 0.8 percent of the ^{90}Sr separated with the Na_2SO_4 as a water soluble phase. The product prepared at 1200°C contained no separate sulfate phase and no water soluble surface deposits of ^{137}Cs or ^{90}Sr .

Approximately 1.0 percent of the cesium volatilized from the 1200°C melt while only 0.4 percent evolved from the 1000°C melt. As shown in Table VIII, the leach rates based on ^{137}Cs were significantly lower when the glass was prepared at 1200°C rather than 1000°C .

In future work, glasses will be prepared from actual sludges and residual liquors as they become available, and also from synthetic sludges containing bismuth phosphate, zirconia, nickel ferrocyanide, and titanate resins.

Glass Melter Development - D. A. Turner

Efforts to procure an automated, continuous bench-scale glass melter were initiated. The objectives of the bench-scale melter test program are to evaluate melter operability under a wide range of operating conditions, to demonstrate that a single continuous electric melter can produce an acceptable glass product from a range of feed compositions, and to obtain materials compatibility information.

The preliminary cost estimate for procurement and installation of the bench-scale melter is \$120,000. Operation is tentatively scheduled for May 1978.

Reconstruction of the mini-melter has been deferred, since this equipment does not meet criteria for a reliable, readily operable laboratory tool.

Radionuclide Removal - G. S. Barney, W. W. Schulz

The objective of this activity is the identification, development, and demonstration of processes for removal of all longlived (half-life > 25 years) radionuclides from Hanford salt cake and residual liquor. The primary goal is to develop technology for partitioning these high-level wastes into a small volume of highly radioactive waste requiring immobilization for long-term or final storage and a large volume of low-level chemical waste that can be safely stored by inexpensive means.

Removal of ^{99}Tc - W. W. Schulz

Conceptual Hanford Radionuclide Removal flowsheets currently provide for removal of long-lived ^{99}Tc from residual liquors and, possibly, from some salt cake solutions. (Typically, residual liquor contains over 100 $\mu\text{Ci/liter}$ ^{99}Tc , while salt cake solutions contain only 3 - 15 $\mu\text{Ci/liter}$ ^{99}Tc .) Strong base anion exchange resins have a high affinity for the TcO_4^- ion, the form in which ^{99}Tc exists in alkaline Hanford wastes. Ion exchange processes and technology for removal of ^{99}Tc from certain Hanford Purex process wastes were developed and tested previously. We anticipate that much of this technology will directly transfer to removal of ^{99}Tc from residual liquors and, if necessary, salt cake liquors. Ion exchange tests completed this quarter with actual salt cake solution confirm this expectation.

In one such test, 208 ml (104 column volumes) of salt cake solution were passed downflow at 25° C and one column volume per hour through a 2.0-ml (0.7 cm diameter) bed of 20 - 50 mesh, NO_3^- form Amberlite IRA 401 anion resin (Rohm and Haas Company). Feed solution for this run was obtained by water

dissolution of a mixture of salt cake obtained from the 102 and 103-SX tanks. It contained, along with other constituents, 5.82M Na^+ , 0.99M OH^- , 0.012M CrO_4^{2-} , $13.8 \mu\text{Ci/liter } ^{99}\text{Tc}$, and $3.36 \times 10^4 \mu\text{Ci/liter } ^{137}\text{Cs}$. Figure 3 is a plot of the concentration of ^{99}Tc in the load cycle effluent versus number of column volumes of feed passed through the resin bed. This plot shows that some 75 to 80 column volumes of feed were passed through the IRA 401 resin before the 10 percent breakthrough point. The breakthrough curve in Figure 3 exhibits the characteristic S-shaped form; the flat early portion of the curve may represent nonsorption of a small amount of ^{99}Tc present in some other species than the TcO_4^- ion.

Following the load cycle, the bed was washed, upflow at 25°C and one column volume per hour, successively with 5.7 column volumes of 1M NaOH , 4.3 column volumes of water, and 7.2 column volumes of 0.25M HNO_3 . These washes removed, respectively, 0.53, 0.37, and 1.19 percent of the ^{99}Tc on the bed. Subsequently, the bed was eluted upflow with 6.7 column volumes of 6M HNO_3 and then washed finally with 9.7 column volumes of water. The 6M HNO_3 solution removed 54 percent of the ^{99}Tc ; another 12.5 percent of the ^{99}Tc was contained in the final water wash. Only 69 percent of the ^{99}Tc loaded onto the bed was accounted for. It is presently believed that the abbreviated volume of 6M HNO_3 eluent was not sufficient to elute all of the ^{99}Tc .

The composite eluent solution contained $86.7 \mu\text{Ci/liter } ^{99}\text{Tc}$, $13.2 \mu\text{Ci/liter } ^{137}\text{Cs}$, 0.0053M Na^+ , and $1.25 \times 10^{-4}\text{M Cr}^{3+}$. These latter concentrations corresponded to decontamination factors for ^{137}Cs , sodium, and chromium of 1.6×10^5 , 7×10^3 , and 617 respectively.

FIGURE 3

BREAKTHROUGH CURVE FOR LOADING OF ^{99}Tc FROM
 SALT CAKE SOLUTION ON IRA 401 ANION EXCHANGE RESIN

Sodium Titanate Development - R. W. Lynch (Sandia Lab.)

Sandia Laboratories continued development and testing of sodium titanate (ST) to be employed as an ion exchanger for removal of strontium in the radionuclide removal process.

A 200-gram sample of sodium titanate prepared by Cerac, Inc., of Milwaukee, Wisconsin, was received and characterized. Table IX summarizes test results on the Cerac material versus proposed specifications and analyses obtained on typical Sandia ST. The Cerac material met all specifications with the exception of titanium to sodium molar ratio. Cerac does not presently have the analytical capability required for sodium determination. The failure to meet the titanium to sodium specification should be resolved when Cerac completes their effort to establish this capability.

In prior work with ST resin using acidic wastes, carbonate formation during resin production (resulting from absorption of atmospheric CO_2 during air drying) was believed to be detrimental due to decreased exchange capacity and the possibility of gas generation. With the switch in emphasis to strontium removal from basic, high-salt waste solutions, the presence of carbonate appears to be an attribute. Data in Table X compare Sr^{++} capacities in neutral solutions and the distribution coefficients (K_d) of strontium in a dissolved salt cake simulant for ST containing varying amounts of carbonate. Also listed in Table X are the strontium capacities of three ST materials, including the Cerac sample, in a dissolved salt cake matrix, 6 N NaNO_3 - 0.6 N NaOH - 0.1 N $\text{Sr}(\text{NO}_3)_2$.

TABLE IX
CHARACTERIZATION OF CERAC SODIUM TITANATE (ST)

	<u>Cerac ST</u>	<u>Typical Sandia ST</u>	<u>Nominal</u>
Weight loss at 700° C	21.57 w/o	20 - 25 w/o	10 - 30 w/o
Titanium content	40.00 ± 0.07 w/o (2)	(1)	(1)
Sodium content	8.47 ± 0.08 w/o (3)	(1)	(1)
CO ₃ ⁼ content	0.25 ± 0.04 w/o (2)	0.2 - 0.5 w/o	0 - 5 w/o
Mole ratio Titanium:Sodium	2.27 ± 0.02	2.00 ± 0.002 w/o	1.92 - 2.08
Strontium capacity	3.73 ± 0.02 meq/g	4.2 meq/g	> 3.5 meq/g

(1) These values are dependent on weight loss at 700° C of a given material in the 10 - 30 w/o loss range given above.

(2) Mean and average deviation from mean in 4 determinations.

(3) Mean and average deviation from mean in 6 determinations.

(4) Mean and average deviation from mean in 3 determinations.

TABLE X

EFFECT OF CARBONATE IN SODIUM TITANATE (ST)

Sample	Drying Conditions	CO ₃ ⁼ (w/o)	Mole Ratio Ti:Na	Sr ⁺⁺ Capacity, Neutral Solution (1)		Sr ⁺⁺ Capacity, ITS Matrix		
				As-Dried	700° C Dry	Kd (2)	(1) As-Dried	700° C Dry
ST - 2A	Vacuum	0.25	2.0	4.17	5.60	1 x 10 ⁵	8.05	10.81
ST - 2A	Air + Vacuum	3.54	2.13	3.84	5.04	1.4 x 10 ⁵	---	---
ST - 2A	Flowing Air	4.93	2.0	4.21	5.53	2.8 x 10 ⁵	8.93	11.73
ST - Cerac	Vacuum	0.24	2.27	3.73	4.76	---	8.00	10.21

- (1) Results are given in meq/g. Values were determined for the as-dried materials, and the 700° C dry values were calculated from weight loss data for the materials.
- (2) These values represent the distribution coefficient for ⁸⁵Sr tracer between a simulated salt cake solution and sodium titanate materials.

In neutral strontium solutions, the presence of carbonate has little effect on strontium capacity of ST, which appears to be more dependent on the relative sodium content. This result is expected as strontium carbonate is relatively insoluble at this pH, and the results do not differentiate between strontium precipitated as the carbonate and that absorbed on the ST.

In the basic dissolved salt cake solutions, both distribution coefficients for ^{85}Sr tracer and overall Sr^{++} capacity appear to increase with increased carbonate content. A surprising result is seen for the Sr^{++} capacities, which are about twice those observed for the same materials in neutral solutions. In addition, the capacity dependence on the relative sodium content is less apparent, perhaps due to sodium ions in the high sodium, basic matrix replacing hydrogen ions at some of the exchange sites and, thus, making them more accessible to Sr^{++} exchange.

A breakthrough experiment was completed using a ^{85}Sr doped, dissolved salt cake simulant including Ni^{+2} , Fe^{+3} , Mn^{+2} , Ba^{+2} , Mg^{+2} , and Ca^{+2} additions as the feed. The column was a 3 cm³ bed of air-dried ST - 2A prepared at Sandia. In addition to breakthrough characteristics, the experiment was designed to look at decontamination factors as a function of flow rate in the regime 2 - 15 column volumes per hour.

The column flow rate profile and effluent ^{85}Sr concentration relative to the feed concentration (C/Co) are plotted versus throughput in column volumes in Figure 4. A breakthrough is observed between 1900 and 2900 column volumes, and the ^{85}Sr appears to increase almost linearly up to a $C/\text{Co} = 0.25$. This value was reached just prior to a weekend, and the small amount of feed remaining dictated a reduction in flow to maintain column operation over the weekend. In the experimental setup used in this work, the column was enclosed in a large polyethylene container to avoid the potential spread of contamination due to equipment failure. Thus, visual inspection was periodic rather than constant. The first of these occurred after 1600 column volumes, at which time the column was colored, i.e., the upper 20 - 25 percent was brown, and the remainder was a light green, indicating the presence of iron and nickel on the ST bed. The quantities of these elements in the 1600 column volumes of feed (~ 8 mg nickel and 12 mg iron) are not sufficient to "load" the column, but were shown in batch experiments to be sufficient to impart the observed color to comparable amounts of ST.

The decrease in the C/Co value coinciding with the decrease in flow rate prior to termination of the experiment is believed to be related to the presence of iron, nickel, and possibly other cations in the ST. Figure 5 shows the column flow rate profile and C/Co values for ^{85}Sr prior to breakthrough. The error bars reflect the standard deviations observed for five 5-minute counting periods on five milliliter samples of effluent, and no apparent dependence of C/Co on flow rate is observed.

FIGURE 4

PERFORMANCE CHARACTERISTICS OF ST COLUMN USING IMPURITY DOPED
SIMULATED DISSOLVED SALT CAKE FEED OVER ENTIRE COLUMN

FIGURE 5

PERFORMANCE CHARACTERISTICS OF ST COLUMN USING IMPURITY DOPED
SIMULATED DISSOLVED SALT CAKE FEED FOR FIRST 2000 COLUMN VOLUMES

During the final several hundred column volumes, the column loading due to stable cations is significant, i.e., the quantity of Fe^{+3} , Ni^{+2} , and Ca^{+2} in the feed throughput totals ~ 3.7 meg. Although the ST capacity for these species in basic matrix is not known, based on the value of Sr^{++} in Table X, it would be conservative to assume 30 percent column loading. The other cations added, Mn^{+2} , Ba^{+2} , and Mg^{+2} , although below detectable limits for atomic absorption, may also serve to increase the loading factor. It is interesting to note that in a similar experiment with a feed containing macro concentrations of stable strontium, described above, breakthrough was observed at approximately a 30 percent loading factor.

The presence of significant amounts of "other" cations sorbed on the ST bed probably contributes to the relative Sr breakthrough volume but does not account for the striking decrease in C/Co with flow rate after the breakthrough. This is probably the result of a change in kinetics due to the column loading, and experiments have indicated the presence of added cations in the salt cake simulant retards the adsorption of $^{85}\text{Sr}^{++}$ on ST in the initial few minutes of contact time.

An identical experiment will be undertaken using a salt cake simulant of the same composition excluding the cation impurities. This is intended to provide both a "best" and "worst" case breakthrough value.

Ozonation of Organic Complexants - W. W. Schulz

That some residual liquors and perhaps also a few salt cake solutions contain organic complexants which interfere with removal of ^{90}Sr (and likely actinides too) has been noted in several earlier quarterly reports. Recent quarterly reports have also presented data which show that ozonation of such solutions at 75-90° C oxidizes these complexants to CO_2 and/or organic materials which do not inhibit removal of ^{90}Sr by precipitation or sorption techniques.

Equipment and technology to determine the total organic content of aqueous radioactive waste solutions became available this quarter and were used to measure amounts of organic material in representative Hanford waste solutions (Table XI). Hanford B Plant wastes are known to contain EDTA (ethylenediaminetetraacetate), HEDTA, (hydroxyethylenediaminetriacetate), and in some cases, citrate; concentrated B Plant wastes, currently scheduled to be stored in double-shell tanks, are expected to contain the highest concentrations of organic complexants of any of the stored Hanford wastes. Salt cake from the 102 and 103-SX Tanks contains small amounts of complexants present presumably in the interstitial liquid; these salt cakes are the only ones of several tested (108-S, 105-S, 105-B, 109-S, and 116-TX) found to contain any complexants. Waste in the 118-TX Tank is of interest because high salt waste from Hanford's Plutonium Reclamation Facility, containing significant amounts of ^{241}Am , is routed to this tank.

TABLE XI

ORGANIC CARBON CONTENT OF REPRESENTATIVE WASTE SOLUTIONS

Waste From Tank No.	Waste Description	Organic Material	
		Total Carbon, <u>M</u>	As EDTA, <u>M</u>
102-SX-103-SX	Salt Cake Solution	0.211	0.0211
110-S	Residual Liquor	1.36	0.136
102-S	Dilute Evaporator Feed (a)	2.63	0.263
118-TX	Z Plant High Salt Waste (b)	3.49	0.349
111-U	Concentrated B Plant Waste	3.96	0.396
106-U	Concentrated B Plant Waste	5.42	0.542

(a) Also contained some concentrated B Plant waste

(b) Also contained some B Plant solvent extraction waste when sample was taken.

In earlier experiments, 540 bed volumes of the solution obtained by water dissolution of a mixture of salt cakes from the 102-and 103-SX Tanks were passed through a 2-ml bed of sodium titanate exchanger. Such treatment uniformly reduced the ^{90}Sr level from 2150 $\mu\text{Ci/liter}$ to 4-5 $\mu\text{Ci/liter}$. Neither passage of the effluent through a second 2-ml bed of titanate nor scavenging with inert strontium phosphate reduced the ^{90}Sr level below 4-5 $\mu\text{Ci/liter}$. Such behavior is consistent with the presence of a small amount of organic complexant (Table XI). Accordingly, to destroy this complexant, a 30-ml portion of the salt cake solution was ozonized (1.86 volume percent O_3 in O_2 ; 2.1 liter/minute) for six hours at 75°C . This treatment yielded about 0.4 ml of black solids and reduced the ^{90}Sr level from 4-5 $\mu\text{Ci/liter}$ to 0.51 $\mu\text{Ci/liter}$.

Other ozonation experiments were performed with waste solutions from Tanks 118-TX and 106-U, both of which contain high concentrations of organic complexants (Table XI). Additional relevant composition data for these waste solutions are listed in Table XII. The concentrations of ^{90}Sr , ^{241}Am , and ^{239}Pu in four-fold diluted, unozonated Tank 118-TX waste were all reduced to 1 $\mu\text{Ci/liter}$ or less by double precipitation of 0.0025M inert strontium phosphate. Ozonation of 30 ml of four-fold diluted Tank 118-TX solutions for six hours at 75° C (1.86 volume percent O_3 in O_2 , 2.1 liter/min.) yielded a small amount of brown precipitate and a solution containing 87 $\mu\text{Ci/liter}$ ^{90}Sr , 18 $\mu\text{Ci/liter}$ ^{241}Am , and 0.40 $\mu\text{Ci/liter}$ ^{239}Pu . A single precipitation of 0.0025M inert strontium phosphate reduced the ^{90}Sr concentration to 0.096 $\mu\text{Ci/liter}$ and the concentrations of both ^{239}Pu and ^{241}Am to below 0.01 $\mu\text{Ci/liter}$.

A large (5 ml) amount of brown and white solids precipitated when 30 ml portions of two-fold diluted Tank 106-U waste (adjusted to 1M NaOH) were ozonated 12 hours at 75° C. The brown solids were identified as $\text{Fe}_2\text{O}_3 \cdot x \text{H}_2\text{O}$ which precipitated when organic ligands such as EDTA and HEDTA were oxidized by O_3 and released complexed Fe^{3+} . Some of the CO_2 produced from ozonolysis of the EDTA and HEDTA volatilized but much of it reacted with NaOH to form Na_2CO_3 . As a result of this latter reaction, the NaOH concentration fell to the point where some of the AlO_2^- precipitated as $\text{Al}(\text{OH})_3$. Washed and dried solids from one 30-ml ozonation experiment weighed one gram and contained 18 weight percent Al, 26 weight percent Fe, and most importantly, about 94 percent of the ^{90}Sr in the diluted Tank 106-U solution.

Ozonation of the diluted Tank 106-U solution for 12 hours at 75° C reduced its total organic carbon content from

5.42 to 1.48M. Ozonated Tank 106-U solution, after dilution with a small volume of 6M NaOH from the solids washing step, contained in one case 170 $\mu\text{Ci/liter}$ ^{90}Sr and in another, 825 $\mu\text{Ci/liter}$ ^{90}Sr . Two precipitations of 0.0025M inert strontium phosphate from the latter solution reduced its ^{90}Sr concentration to 100 $\mu\text{Ci/liter}$. Precipitation of strontium phosphate from the former ozonated 106-U solution did not reduce its ^{90}Sr level below about 140 $\mu\text{Ci/liter}$.

TABLE XII

COMPOSITION OF SOLUTIONS FROM
TANKS 118-TX and 106-U^a

<u>Component</u>	<u>Tk 118-TX</u>	<u>Tk 106-U</u>
Na^+ , <u>M</u>	8.46	NA ^b
NO_3^- , <u>M</u>	3.47	4.86
OH^- , <u>M</u>	1.16	0.95
NO_2^- , <u>M</u>	2.0	1.25
AlO_2^- , <u>M</u>	0.78	0.90
CO_3^{2-} , <u>M</u>	0.50	0.49
CrO_4^{2-} , <u>M</u>	0.13	NA
Fe^{+3} , <u>M</u>	0.0033	0.030
^{137}Cs , $\mu\text{Ci/l}$	3.51×10^5	3.30×10^5
^{90}Sr , $\mu\text{Ci/l}$	7.28×10^5	1.32×10^5
Total Pu, $\mu\text{Ci/l}$	1.6	NA
^{241}Am , $\mu\text{Ci/l}$	66.8	NA

^aOrganic carbon content shown in Table XI.

^bNA = not available

Hot Cell Pilot Plant Tests

An extensive series of hot cell tests to demonstrate radionuclide removal flowsheets and technology with kilogram amounts of salt cake and liters of residual liquor is planned for FY 1978 and FY 1979.

Laboratory work next quarter will be concentrated on further tests with concentrated B Plant aqueous wastes (e.g. Tank 106-U liquor) containing large amounts of organic complexants to show that a combination of ozonation and scavenging (or titanate sorption) will reduce ^{90}Sr levels to or below $1 \mu\text{Ci/liter}$. Efforts will also continue to acquire equipment and materials needed for hot cell pilot plant scale radionuclide removal tests. We also plan to complete current ^{99}Tc removal tests and to start flowsheet tests to remove radionuclides from the salt cake solution obtained by dissolving salt cake from Tank 109-S.

Fractional Crystallization - G. S. Barney

Separation of inert (non-radioactive) salts from radionuclides in stored wastes by fractional crystallization was studied further. Nitric acid was added to the solution of waste salts in order to achieve supersaturation of NaNO_3 . According to published data, the solubility of NaNO_3 decreases greatly as the concentration of HNO_3 in solution increases (see Figure 6). Therefore, the yield of NaNO_3 crystals should be much greater when nitric acid is added to the mother liquor.

FIGURE 6

SOLUBILITY OF NaNO_3 IN HNO_3 SOLUTIONS FROM
W. F. LINK, "THE SOLUBILITIES OF INORGANIC AND
METAL-ORGANIC COMPOUNDS, VOLUME II, p. 1072

Yields of NaNO_3 crystals and decontamination factors for Sr and Cs were measured for several recrystallizations of a synthetic salt cake waste solution in the presence of HNO_3 . A solution saturated with NaNO_3 at 100°C (spiked with ^{85}Sr and ^{137}Cs) was made 6M in HNO_3 and cooled to room temperature. The crystalline product was filtered, washed with 7M HNO_3 and dried. This product was then recrystallized several additional times. Decontamination factors and crystalline yields are given in Table XIII for each crystallization.

The data show that crystallization from a nitric acid solution greatly improves the separation. Yields are about four times larger and decontamination factors are about ten times greater than when using water as the crystallization medium. Only one or two crystallizations are necessary to reach the goal of $<1 \mu\text{Ci/liter}$ of ^{137}Cs or ^{90}Sr in a saturated solution of the product.

A conceptual flowsheet for radionuclide removal by fractional crystallization from nitric acid solutions is shown in Figure 7. The washing-recrystallization column allows continuous fractional crystallization and washing of the crystals. Several designs are available* and have been used on a large scale. Multistage separations can be obtained with columns of this type so that a very pure NaNO_3 product can be obtained.

* G. J. Arkenbout, "Continuous Fractional Crystallization", Chemtech, 6, 596-599 (1976).

TABLE XIII
CRYSTALLIZATION YIELDS AND DECONTAMINATION FACTORS

	<u>Decontamination Factors</u>		<u>Yield, Percent of Original Salt Cake</u>
	<u>^{85}Sr</u>	<u>^{137}Cs</u>	
1st Crystallization	302	510	68.3
2nd Crystallization	> 490	> 3820	45.5
3rd Crystallization*	—	—	31.4

* ^{85}Sr and ^{137}Cs concentrations were too low to detect in mother liquor and crystals.

FIGURE 7

CONCEPTUAL FLOWSHEET FOR FRACTIONAL CRYSTALLIZATION
SEPARATION OF NaNO_3 FROM RADIONUCLIDES IN HNO_3 SOLUTION

Solvent Extraction - G. S. Barney

Separation of sodium salts from radionuclides by solvent extraction was studied using a new solvent. 1, 1, 2, 2-tetrachloroethane was used as a diluent for the t-butylbenzo-15-crown-5 extractant because of its ability to dissolve large amounts of the crown ether and its high density. The goal was to obtain sodium extraction coefficients large enough for practical application of counter-current extraction.

The extractant used was 1.54M t-butylbenzo-15-crown-5 in 1, 1, 2, 2-tetrachloroethane. Extraction coefficients (E_a^0) for Na, Cs, and Sr were measured for a range of Na concentrations in the aqueous phase. The results are shown in Figure 8.

Sodium is extracted very strongly by this solvent. Sodium concentrations as high as 1.2M were measured in the organic phase. Values for sodium E_a^0 increase as the aqueous phase concentration of Na decreases. However, separation of Cs is good only at high concentrations of Na because of competition between Na and Cs for the organic extractant. Separation of Sr is very good using this extractant. Strontium extraction coefficients of < 0.005 were observed.

4.4 STORAGE AND DISPOSAL ALTERNATIVES - D. A. Turner

The objectives of this program are to:

- o Monitor Office of Waste Isolation programs for development of bedded salt and basalt final storage technology, and develop compatible waste containerization, handling and transportation technologies.

FIGURE 8

EXTRACTION COEFFICIENTS FOR Na, Cs, and Sr
USING 1.54M T-BUTYLBENZO-15-CROWN-5 IN
1, 1, 2, 2-TETRACHLOROETHANE AS EXTRACTANT

- Identify and develop technology for long-term or final storage of immobilized Hanford high-level waste in near-surface engineered storage facilities.
- Develop economic and technical criteria against which alternative waste storage concepts may be compared and evaluated.

Kaiser Engineers continued their evaluation of the technical feasibility of alternative surface and geologic waste storage options (refer to Section 4.5, Alternatives Studies and Assessments). Program effort during the report period was devoted to support of this activity.

4.5 ALTERNATIVES STUDIES AND ASSESSMENTS - R. Y. Lyon

The objectives of this program are:

- To conduct a comprehensive analysis of all reasonable alternatives being considered for safe long-term management of Hanford defense high-level wastes,
- To prepare technical documentation of the analysis for public review and comment (Hanford Defense High-Level Waste Document),
- To document for public review and comment the environmental impact of all alternatives being considered (Generic Environmental Statement),
- To document the environmental impact of the alternative selected for implementation (Project Specific Environmental Statement) at the time construction funds are requested.

Similar documents are being prepared at the Idaho National Engineering Laboratory and the Savannah River Plant as part

of an overall ERDA program to keep the public informed of planning for long-term management of defense high-level wastes at the three sites.

Analysis of Alternatives

The objectives of this activity are to identify all reasonable alternatives for long-term management of the Hanford defense high-level wastes; determine the technical feasibility, advantages and disadvantages, and capital and operating costs for each alternative; conduct safety and risk analyses for each alternative; and determine resource requirements. This phase of the work is scheduled for completion by February 15, 1977, to permit integration of results into the Defense Waste Document. The analyses will be continued to provide information of the Generic Environmental Statement.

The waste management options to be analyzed were reconsidered during the report period. They now consist of 25 options for the salt cake, sludge, and residual liquor; two options for the decontaminated salt product of the radionuclide removal process; and eight options for the ^{90}Sr and ^{137}Cs capsules. They reflect the addition of three storage options for the clay calcination product without radionuclide removal and a decrease of one decontaminated salt storage option. The options are shown in matrix form on Figure 9 and graphically on Figures 10 and 11. The alpha-numeric codes shown on the rectangular boxes of Figures 10 and 11 will be used to identify unique cost centers. Different identifiers for seemingly identical elements result mainly from the different volumes of waste product which must be handled. This is particularly true for the columns headed Containerization, Transportation, and Terminal Storage Mode.

WASTE PRODUCT		TERMINAL STORAGE MODE			
		DEEP GEOLOGIC STORAGE		SURFACE STORAGE	
		BASALT CAVERN ON-SITE	SALT CAVERN OFF-SITE	VAULT ON-SITE	WASTE TANK ON-SITE
SALT CAKE, SLUDGE AND RESIDUAL LIQUOR	GLASS ¹	1, 2 ¹	3	4, 5 ⁴	
	CLAY CALCINATION ¹	6	7	8	
	CEMENT ¹	9	10	11	
	BITUMEN ²	12	13	14	
	UNPROCESSED ¹ WASTE	15			16 ⁵ , 17 ⁶
CAPSULES	EXISTING CAPSULES WITH OVERPACK	A	B	SEALED STOR. CASK C	WATER BASIN D ⁵ , E ⁵
	CAPSULES CONVERTED TO GLASS	F	G	SEALED STOR. CASK H	
SALT	DECONTAMINATED ¹ SALT	I		SURFACE TRENCH II	

- Notes:
1. Denotes waste product options evaluated with and without radionuclide removal.
 2. Denotes waste product options evaluated with radionuclide removal only.
 3. Denotes waste product options evaluated without radionuclide removal only.
 4. Denotes bulk storage options.
 5. Denotes options with solidification of residual liquor, engineered improvements to existing facilities, and overpacking of capsules.
 6. Denotes options with no solidification of residual liquor, no engineered improvements to existing facilities, and no overpacking of capsules.

Revised
12/13/76

FIGURE 9
KEY TO TECHNICAL OPTIONS
FOR LONG-TERM MANAGEMENT OF HANFORD DEFENSE HIGH-LEVEL WASTE

RE 10

TECHNICAL OPTIONS
E HIGH-LEVEL WASTES
and Residual Liquor)

FIGURE 11
FLOW SCHEMATIC OF TECHNICAL OPTIONS
FOR HANFORD DEFENSE HIGH-LEVEL WASTES
(Capsules and Decontaminated Salt)

For purposes of this analysis, the assumption has been made that implementation of the alternative selected for long-term management will start in FY 1990 and continue for 15 years.

During the report period, ARHCO personnel completed preparation of flowsheets and accumulation of other source information required for technical feasibility and design studies, and risk analysis of the options under evaluation. Flowsheets were prepared for Clay Calcination Immobilization, Cement Immobilization, Bitumen Immobilization, Decontaminated Salt Treatment and Disposal, and Glass Immobilization of Strontium and Cesium from Capsules. Sketches and narrative information were prepared for layout and equipment requirements in a basalt cavern, overpacking of capsules for surface and geologic storage, and for conversion of encapsulated cesium chloride and strontium fluoride to glass, with subsequent storage in a sealed storage cask at ground surface. Work during the next quarter will involve development of any basic information needed for design or risk analysis studies, but the major emphasis will be on preparation of the Hanford Defense Waste Document.

During the report period, Kaiser Engineers (KE) continued evaluation of the technical feasibility of the various options and processes under consideration, performed miscellaneous safety analyses, and initiated estimation of capital and operating costs. Costs estimates for three alternative waste retrieval concepts have been developed: (1) a sluicing and slurry-pumping system based upon designs already in existence or currently employed in waste management operations, (2) a retrieval system based upon using a mechanical arm as the primary means for positioning specially designed tools and equipment in the waste tank for retrieval of waste, and (3) a retrieval system based upon a creative approach developed by Kaiser Engineers.

Kaiser Engineers will prepare an independent report of its studies as shown below:

- Preliminary drafts for each task As completed
- Incomplete first draft February 15, 1977
- First complete draft April 1, 1977
- Final report To be specified

In October 1976, Nuclear Services Corporation (NSC), Campbell, California, was retained on an ERDA prime contract to perform the risk analyses required as part of the overall evaluation of alternatives. Efforts by NSC personnel during October and November were directed toward digesting the background information supplied by ARHCO and becoming familiar with the options to be analyzed. A review of NSC efforts was held on December 9, 1976 at NSC headquarters. Preliminary accident scenarios were presented and items of required technology were identified. In order to maintain continuity at Hanford, radiation doses to be used in the analyses will be calculated by computer programs developed by Battelle-Northwest. Familiarization of NSC personnel with the programs was conducted by BNW. A population density and distribution study for a 90-mile radius centered on the Hanford Meteorological Tower and for the period from 1980 to 2300 was completed by BNW during the report period. During the next quarter, NSC will complete the risk analyses and prepare draft sections for the HDWD by February 15, 1977.

Hanford Defense High-Level Waste Document

The objective of this activity is to document the results of the analysis of reasonable alternatives being considered for safe long-term management of Hanford defense high-level wastes. A draft document is to be prepared for ERDA review by April 1, 1977, with issue of the final document by ERDA scheduled for September 1977.

The Hanford Defense High-Level Waste Document (HDWD) will be one of three similar documents being prepared for the Idaho National Engineering Laboratory, the Savannah River Plant, and Hanford. A final draft of the Savannah River (SR) document, which is to be the guide for format and content, was reviewed in December, and comments were submitted to ERDA-RL for transmittal to Savannah River. The HDWD will be less detailed than the SR report, particularly in the area of risk analysis, since the basic information is not available and there is not enough time to complete the exhaustive studies that were done at SRP. A detailed outline of the HDWD was prepared, writing assignments were made, and some writing started. During the next quarter, preparation of the draft HDWD for ERDA review and comment will be completed.

Generic Environmental Statement (GES)

The objective of this activity is to prepare documentation on the environmental impact of all alternatives being considered for long-term management of the Hanford defense high-level wastes. Issue of the document for public review and comment is presently scheduled for December 1978.

No work has been started on this activity, although it is expected that many of the results of the analysis of alternatives now in progress will be applicable to the GES. The main effort will be started when the HDWD has been essentially completed.

5.0 LONG-TERM MANAGEMENT OF CONTAMINATED SOILS AND SEDIMENTS

The objective of this program is to establish the definitive data base and demonstrated technologies required for the selection, establishment, and implementation of a program(s) that provide efficient, effective, and socially acceptable management of the radionuclides stored in or on the soils and sediments in the 200 Areas' control zone.

The planning, organizing, and integration required to establish a coherent technology program that will meet the stated objective has proceeded since the start of fiscal year 1977. The program plan will integrate ongoing work with activities specifically established for this program. Also, the program overlaps and interfaces with the High-Level Waste and Surplus Contaminated Facilities programs, thereby requiring integration and coordination with those programs.

A first draft of the Technology Program Plan is scheduled for the next quarter.

5.1 CONTAMINATED SEDIMENTS CONTROL

Site Characterization General - K. R. Fecht

The activity objectives are: (1) to characterize various intermediate- and low-level fission product waste disposal facilities with respect to the fission products and stratigraphy, and (2) to determine the stability of the fission products in the sediments at waste disposal facilities.

The U. S. Testing Company has analyzed 970 sediment samples for size and calcium carbonate analysis during the reporting period. The sediment sample analysis aids in developing geologic maps of the stratigraphy beneath intermediate-level waste disposal facilities.

The geophysical logging of 240 crib wells is 90 percent complete (gamma radiation logging, 100 percent; neutron logging, 80 percent; and density logging, 100 percent). Selected gamma radiation profiles from crib wells logged between 1958 and 1970 are being digitized (60 percent complete). The older and current profiles will be used to determine the stability of radionuclides in the ground beneath crib sites. The gamma radiation, neutron, and density profiles will be used in conjunction with geologic maps and cross sections to locate the lithologic units where contamination is found.

Trench Characterization - S. M. Price, M. M. Kiilsgaard

The objectives for Actinide Trench Characterization are: (1) reactivity assessment - locate probable zones of greatest actinide concentration within trenches and assess the nuclear reactivities of these zones, (2) actinide distribution - delineate the distribution of actinides stored within trenches as a function of the structure, stratigraphy, and mineralogy of the underlying sediments, (3) actinide stability assessment - assess the migration potential of actinides stored within trenches in order to determine the immediacy for implementing waste storage alternatives and, (4) drilling and sampling technology - design, develop, and test containment facilities and operational techniques for the retrieval of actinide-bearing sediments and wastes.

Reactivity Assessment. The results of pulsing of test holes located at the initial points of waste distribution within the 216-Z-1A(c) and 216-Z12 Trenches (Figure 12) were reported by Battelle Northwest. Neutron density profiles for each test hole indicated that the neutron count rate was above background only from one to five feet below the bottom of each trench. Pulsed neutron source measurements of these zones revealed no appreciable reactivity at either site. It was concluded that the sediment systems pulsed at each site were so far subcritical that K_{eff} calculations were not practicable. The K_{eff} was arbitrarily reported as being less than 0.5 for each system.

Actinide Distribution. The ^{238}Pu + ^{239}Pu and ^{241}Am content of samples submitted from wells W18-152 and W18-153 (Figure 12) were determined by LFE Laboratories. The actinide values for these samples to a depth of approximately 109 feet below the ground surface are listed in Table XIV. The analyses show that the plutonium content of the samples is <10 pCi/g of sediment (a suggested maximum permissible concentration for plutonium in sediments) except for samples collected near the bottom of the trench, ~21 feet below the ground surface. The results also reveal that the east-west lateral distribution of plutonium and americium near the head of the 216-Z12 Trench is predominately restricted to a 23 feet wide zone either side of the central distributor pipe.

Autoradiographs of an "undisturbed" sediment sample from 15.5 feet below the bottom of the 216-Z9 Trench indicated that the bulk of the plutonium is associated with glassy-looking fragments. The composition and crystal structure of such fragments was examined during the report period by

FIGURE 12

Z-TRENCH COMPLEX WELLS

TABLE XIV

ACTINIDE DISTRIBUTION IN WELLS
W18-152 AND W18-153 (Fig. 12)

	pCi $^{238}\text{Pu} + ^{239}\text{Pu}$ /gm <u>Sediment</u>	pCi ^{241}Am /gm <u>Sediment</u>
<u>W18-152</u>		
14' *	0.79	11.0
25' **	22.5	5.8
33'	0.04	3.5
44'	0.35	0.82
55'	0.38	1.9
70'	7.7	2.9
86.5'	0.84	3.3
98'	0.034	0.88
108.5'	0.92	2.9
<u>W18-153</u>		
21' **	123	51
30'	4.6	103
40'	1.2	2.4
50'	0.54	2.0
60'	1.2	1.7
80'	0.2	0.58
88'	0.45	1.12
94'	0.37	2.2
104'	0.95	1.9
109'	0.44	1.4

* Background Sample

**Samples Collected at Approx. Same Depth as Bottom of
216-Z-12 Trench

Battelle Northwest personnel. X-ray diffraction work revealed that these fragments are nontronitic clays of the smectite group. The work further showed that the clays are less crystalline than the standard nontronites, a structure attributable to exposure to the acidic environment of the 216-Z-9 Trench. Electron microprobe analyses showed the presence of relatively large amounts of phosphorus in the clay indicating that the nontronites either absorbed an organic-plutonium complex or absorbed organics in an unrelated process. In any case, the nontronitic clays, which existed in the sediments prior to trench utilization, apparently functioned as a highly-absorbent media for plutonium.

Actinide Migration. A report summarizing a sequential leaching technique that could be used to assess environmental conditions under which actinide migration in trench sediments could occur has been completed by Battelle Northwest personnel. A review of the report revealed that insufficient work was carried out to allow an adequate assessment of the applicability of the technique. Further work has been postponed until the program direction can be reevaluated.

Analysis of large-volume water samples from three wells in the vicinity of the 216-B-5 Reverse Well - Holding Tank - Tile Field Complex (Figure 13) (Pu inventory ~5Kg) has been completed. The results of the analyses show that values of $^{239}\text{Pu} + ^{240}\text{Pu} + ^{238}\text{Pu}$ for the samples are near fallout levels in sea water, well below mpc levels of Pu for water in an uncontrolled area (Table XV). The presence of ^{238}Pu in the samples strongly suggests that the plutonium present originated from fallout. The plutonium introduced into the reverse well was from the first plutonium production.

FIGURE 13

216-B-5 REVERSE WELL-HOLDING TANK - TILE FIELD COMPLEX

TABLE XV

$^{239+238}\text{Pu}$ and ^{238}Pu Content of Water Sampled
From Three Wells in the Vicinity
of the 216-B-5 Reverse Well
(dpm/1000 liters)

Well	^{238}Pu	$^{239+240}\text{Pu}$	Total Pu
E-28-5	0.39	1.74	2.13
E-28-1	0.32	1.18	1.5
E-28-7	4.7	305.5	310.2
Sea Water	.01-.02	0.3-1.0	
MPC*			13.3×10^6

*ERDA Manual Chapter 0524

It did not have any ^{238}Pu and contained very low, less than 1.5 percent, ^{240}Pu . No significance can be ascribed to the highest plutonium levels being obtained from the sample taken closest to the reverse well.

Sediment Stabilization - P. J. Wiater, G. M. Holter

The objective of this activity is to establish the technologies that provide the capability to stabilize surface and subsurface contamination and to establish criteria for the stabilization of contaminated sediments.

A field test on grouting a well with five different matrices was conducted in 1976. Laboratory tests on well grouts by Battelle Northwest showed that asphalt, or cement and asphalt, would be good barriers to prevent movement of activity to the groundwater. However, the field test indicated that further development of injection techniques is necessary before a satisfactory seal can be established using asphalt-cement grout.

The Battelle Northwest study to characterize liquid laundry effluents and evaluate treatment options for the effluents is continuing under ARHCO direction.

Subsurface Stabilization - P. J. Wiater

Subsurface Bio-Barriers. The original design (BNWL-2035) of the S-11 Culvert Bio-barrier Study has been changed. The four bentonite control culverts have been eliminated and a new barrier added to these culverts. The new barrier is a combination of urea borate and asphalt. Early results have shown that roots can pass through an asphalt barrier but not a urea borate barrier. The combination of the two may provide a barrier that would resist root penetration over a longer time period.

Subsurface Grouting

A grouted moisture barrier, in the form of a tank, was formed by injection grouting into the 200 West Area sediments. The sediments were removed from the center of the tank and the resulting tank leak tested by filling it with water. A leak in the bottom of the grouted tank caused the water to drain out at a rate of one gallon per hour. Further investigations are necessary to determine why the grout did not form a sealed container.

Surface Stabilization and Cleanup - G. M. Holter

Surface Stabilization. A laboratory test was initiated to determine the effect of soil affixants on the growth of cheatgrass and Russian Thistle. The parameters to be measured include seed germination, seedling emergence, and subsequent plant growth. Information gained here will aid development of stabilization techniques for contaminated zones. Stabilization is an effective method for reducing radionuclide resuspension in such zones.

A herbicide study will be initiated in the spring to determine if Russian Thistle growth can be selectively controlled.

Vacuum Cleanup Systems - G. M. Holter

Work on the mini-vacuum system is continuing. Several operational problems have been identified and resolved. Work has been slowed due to delays in equipment delivery.

Liquid Laundry Effluent Control - G. M. Holter

The liquid laundry effluent characterization study was completed in October 1976, and a report was issued. The report included recommendations for treatment of the effluent. These recommendations will be tested in a pilot-plant study which is now underway. Large liquid laundry-effluent samples are being collected and run through a water treatment pilot-plant facility. This study will also provide additional information to be used in design of a full-scale treatment facility.

This study is progressing on schedule, and is expected to be completed in March 1977.

6.0 ENVIRONMENTAL IMPACT TECHNOLOGY

6.1 HYDROLOGY

Groundwater Hydrology - R. A. Deju

This activity is aimed at obtaining the hydrologic data needed to characterize the Hanford groundwater flow regime, developing predictive models of groundwater flow, and developing management and control strategies to avoid groundwater contamination.

Water table measurements were taken in December 1976 and will be interpreted during January 1977. A map of the water level of the unconfined aquifer will be issued at that time. Information obtained from the water level monitoring program has been filed in our computer data bank.

Plans for drilling six research wells in the unconfined aquifer to test the hydrologic properties were laid out during fiscal year 1976. These plans were to be implemented during fiscal year 1977, but they have been postponed for a year due to lack of funding. Pump tests throughout the aquifer will be conducted at six existing sites to assess the aquifer's hydraulic conductivity and porosity. Three wells will be drilled into the deeper confined aquifers and pump tests conducted to assess the hydrologic properties of these aquifers.

As part of the geophysical logging program, the physical and hydrologic properties of selected core well samples were correlated with log responses. A range of hydrologic and physical properties are being assigned to each lithologic unit found to be significant in basinwide groundwater movement. A report is being prepared on the development of geophysical correlation techniques that can apply to the unconsolidated sediment underlying the Hanford Reservation.

Work continued in constructing preliminary isopack maps of the formations underlying the Hanford Reservation. Maps of the top of the basalt, the bottom of the unconfined aquifer, and the sequences of clay layers are being prepared using driller logs and core sample descriptions.

In the mathematical modeling area, work was conducted to provide the methods necessary to extend arrival distribution concepts to practical Hanford hydrologic problems. It is expected that application for such functions in a man-computer interactive system will permit rapid preparation of environmental assessments. A program has been developed to allow pump test data to be entered into the computer data file and a full hydrologic evaluation to be conducted by the computer. Work also continued on the evaluation of the Gupta three-dimensional groundwater model and the assessment of its application to Hanford.

The document, BNWL-2127, "MMT-DPRW Transport Model: Theory and Numerical Implementation," November 30, 1976, was received for review and will be published upon completion of the reviewing processes. All work to date is on schedule and all milestones during the quarter were met.

Offsite Migration - R. A. Deju

This activity is aimed at determining the concentration and ultimate fate of any radionuclides that may be present or could potentially contaminate the confined aquifers underlying the Hanford Reservation. This information is needed to assure isolation of Hanford radioactive wastes from any users of water from the confined aquifer outside of and adjacent to the Hanford Reservation.

A study to characterize the lithologic and hydrologic properties of interbeds found between flows of the Columbia River Basalt Group was completed during the quarter. A draft of the report documenting the study has been reviewed and will be issued by February 28, 1977. This study concluded that these interbeds are extremely clayey and variable in their hydrologic properties. Geochemical information on the interbeds cannot be used for correlation purposes. Plans were drafted to sample additional wells in the confined aquifers and examine the potential for migration of contaminants offsite through these aquifers.

The first sampling period was in late December 1976. These samples will be analyzed and reported subsequently. The analytical program includes a series of quality assurance constraints. All work to date is on schedule and all milestones during the quarter were met.

Radionuclide Migration in Groundwater - R. A. Deju

This activity is dedicated to definition of the main channels of radionuclide migration through the groundwater pathway and to reduction of radionuclide transport throughout these channels. Such objectives are obtained by conducting geological, geophysical, and geochemical studies.

The gravity program is continuing to determine the shape and location of the ancestral buried channels of the Columbia River. All gravity data collected during the summer of 1976 have been compiled as part of the three-dimensional modeling effort to plot the shape of the uppermost basalt surface and determine the depth to bedrock. A report on two-

dimensional Talwani models is being completed. This report will discuss the variations in the anticlinal-synclinal framework of the Reservation and should point out areas of low density outwash which behave as rapid-traveling regions for radionuclide transport.

A contract has been prepared and released in December 1976 to prepare a report examining the three-dimensional variation in hydrologic properties of the Hanford groundwater regime.

Four sediment samples have been selected for use in the radionuclide sorption experiments. The samples were taken from cores in the saturated zone of two wells (299-W11-26 and 699-54-176). They can be classified as silty, sandy gravel; calcareous, sandy silt; and sandy silt. The sediment samples have been screened and separated into gravel, sand, and silt-clay fractions. Duplicate samples of each have been mounted in epoxy resin and polished for electron microprobe examination.

Laboratory equipment for experiments has been ordered. All work to date is on schedule and all milestones during the quarter were met.

Long-Term Groundwater Management - R. A. Deju

This activity is aimed at identifying potential ways in which high-level wastes can come in contact with groundwater, and examining the potential impact to the groundwater regime of past and projected activities in the high-level waste storage areas.

Plans for the establishment of piezometers throughout the Hanford Reservation have been completed and contracts for their installation have been put out for bids. These

piezometers furnish the means of detecting vertical, as well as horizontal, contamination from the 200 Areas. A study entitled "Geohydrologic Study of the West Lake Basin" summarizes the extent that the unconfined aquifer is affected by waters from Gable Mountain Pond and B Pond. This study was issued in October 1976. A similar study is being prepared for U Pond. Mathematical models are being used to assess the environmental impact of these facilities.

A contract was released for pump testing wells in the Cold Creek Valley. This program is aimed at determining the amount of recharge into the Hanford Reservation throughout the confined aquifer that enters through the Cold Creek area. All work to date is on schedule and all milestones during the quarter were met.

6.2 TRANSPORT

Moisture Transport Studies - P. G. Easley, G. V. Last

The objectives of this activity are: (1) to characterize moisture transport in partially saturated sediments in the 200 Areas Control Zone, (2) to formulate mathematical equations accurately describing the nonisothermal flow of liquid, vapor, and gas through porous media, and (3) to relate moisture movement rates and direction of movement to radionuclide transport.

Field Studies of Moisture Movement. Measurements of moisture distribution in the ground have continued at the lysimeter site. No new moisture front has appeared this fall, due to the lack of rainfall, and the moisture envelope from the previous rainy season has still not completely dried out. Equipment breakdowns delayed some readings, but repairs were completed and a new electronics package was purchased so that there will be a backup.

Moisture profiles from the last two years have been compiled and summarized in the 1974-1975 Water Years Report. A final draft has been sent out for comment.

Purchasing of readout equipment and control devices for the sensors in the lysimeters is nearly complete. An estimate has been received from J. A. Jones for the installation of the equipment.

Mathematical Modeling. The development of a mathematical model to simulate moisture movement has been nearly completed in the one-dimensional form. A paper describing the model and some results, along with comparisons with field data, was presented at the Fall Annual Meeting of the American Geophysical Union. Reviews of the paper were received from researchers in the field. There are still minor problems in the computer code related to numerical difficulties that occur when dry soil is simulated.

Properties of Sediments. Determination of the hydrologic properties of lysimeter sediment was done by two researchers, as a quality check. These measurements were used in the computer model to simulate lysimeter sediment. Also, measurements of saturation versus water potential were taken over a range of temperature from 6 ° to 30 °C. A small but distinct dependence of temperature was found, and these results will permit better quantitative determinations of temperature effects on the parameters used in the computer model.

Tank 106-T Leak Investigation. Assistance was provided in the update of Tank 106-T Leak Investigation. The monitoring wells were logged with a neutron moisture probe and moisture profiles were made and plotted to determine the potential for further downward movement of moisture from the tank leak.

Biotic Transport - V. A. Uresk

Identification of biological mechanisms acting as vectors for transporting radionuclides to man is necessary to ensure that control of waste radionuclides is maintained within environmentally acceptable limits.

Radioecological Characterization of Waste Disposal Sites - V. A. Uresk

These studies are designed to provide baseline radioecological information about the plants and animals inhabiting the 200 Areas so that their actual and/or potential role in the release, concentration or transport of radioactive wastes can be assessed more completely.

A status report summarizing current knowledge concerning ecosystem functioning, environmental impact and the importance of the biotic component to waste management operations is in progress. The report will be a synthesis of information derived from studies conducted in the 200 Areas, the Arid Lands Ecology (ALE) Reserve, other nuclear sites, and the pertinent literature which has application toward understanding and solving problems related to biological interactions with radioactive waste management. First draft completion is scheduled for early February.

Analytical results from a two-year study of coots inhabiting the waste ponds in the 200 Areas showed that the average concentration of ^{137}Cs in the breast muscle was 722 pCi/g dry weight. A person consuming the muscle from one coot per day would after six months reach an equilibrium level of 1.8 μCi total body burden or 61 percent of the maximum ^{137}Cs body burden for nonradiation workers (3 μCi TB) and 1.8

times greater than that for the general public ($1 \mu\text{Ci}$) as recommended by the International Commission on Radiological Protection. To put this into the proper perspective, there are about 300 coots using the ponds annually, while in the entire state there are about 150,000 coots. Approximately 8,000 of these are shot yearly by hunters. Therefore the total amount of ^{137}Cs introduced to food chains in Washington would be $0.3 \mu\text{Ci/year}$. This amount would be distributed among the 74,000 hunters in the state each year (4 pCi/individual) with a resulting dose of $1.7 \times 10^{-7} \text{ rem/person}$.

Seven burial grounds were sampled previously to determine the pattern of plant succession with respect to soil characteristics, stoniness, burial ground age and radionuclide concentration in vegetation. The burial grounds sampled included 216-S-4, 216-A-6, 216-A-9, 216-B-36, 216-S-22, 218-E-8, and 218-E-2. Results show that the cheatgrass cover increases with burial ground age and that recently covered sites have significantly more tumbleweeds. The 216-A-6 Site (1970) had more than twice the tumbleweed cover found on the other sites and was also the stoniest. The 216-S-4 (1956) had the fewest tumbleweeds, but a high occurrence of rabbit-brush, a deep-rooted, long-lived shrub. The 216-S-22 Site was the least stony site.

As part of a study determining radionuclide concentrations and dose rates to small mammals near U-Pond, three trap sessions were undertaken with a total of 50 mice captured. Twenty of these were recaptures from previous trapping sessions with 18 of these 20 carrying dosimeters.. Results from the dosimeter analysis are not yet available. Future plans include radiochemical analysis of the recaptured animals for comparison with exposure as determined by dosimetry.

Results from a study of the radioecology of Hanford birds of prey show that the Great Horned Owl feeds primarily on small rodents (especially pocket mice), but also preys upon rabbits, birds (especially magpies), reptiles, insects, fish, bats and scorpions. Past studies have occasionally revealed contaminated prey items in the pellets and castings of local birds of prey. Samples of food items from the castings, eggs and tissues from dead raptors are being analyzed for radionuclides, heavy metals and pesticides. Maps showing all of the locations of raptor nests on the reservation have been prepared. Although several nests occur within the exclusion areas, the majority are found in undeveloped areas of the reservation.

Laboratory Studies of Factors Affecting Biotic

Uptake - V. A. Uresk

These studies are designed to provide quantitative information concerning the effects of variables such as soil properties, waste chemical form, microbial action, weathering effects, plant structure and nutrient requirements on plant and animal uptake of radionuclides. The information will be used in the formulation and maintenance of environmentally sound waste management practices.

A growth chamber study of the effects of soil properties and waste composition on plant uptake of strontium and cesium from five Hanford soils was recently completed. The concentration ratio values for cheatgrass grown on the five different soils ranged from 0.0154 to 0.0641 and 3.54 to 12.29 for cesium and strontium, respectively. The results for tumbleweed were similar with concentration ratios ranging from 0.0078 to 0.0662 and 4.44 to 15.99 for cesium

and strontium, respectively. Greatest plant uptake was from the Burbank and Rupert soils which are those most common in the 200 Areas. Nonradioactive elemental analyses were performed on both the five soils and the plant tissues to provide a basis for evaluating the effect of elemental analogues on the soil transport and plant availability of radionuclides. The analogues being determined are rubidium, potassium, barium, magnesium, calcium, cesium and strontium. Also the physiochemical properties of the soils, such as organic matter, particle size distribution, mineralogical properties, soil pH, and extractable radioactive and non-radioactive cesium and strontium are being determined. Extractants being used include ammonium oxalate, calcium chloride, ammonium acetate, and water. These parameters for each soil are being studied and statistical correlations performed to determine those soil factors which influence the bioavailability of strontium and cesium.

Resuspension Studies - R. D. Fox

The Resuspension Study activity will provide a wind field model to predict site movement of radionuclides through the air. Source terms will be established for the model through actual measurements, data from other programs, and models where appropriate.

A draft report on resuspension was reorganized and edited. The document is currently in the final stages of review and will probably be issued during the next quarter. Work during the balance of the year will be limited to program review and document preparation.

6.3 SEISMOLOGY AND GEOLOGY

Tank Farm Geology - W. H. Price

The objectives of this activity are to: (1) document the structure and stratigraphy of sediments underlying the 200 Areas Control Zone for use in contaminated soil and sediment management, (2) identify significant hydrological properties of each stratigraphic unit for evaluating the impact of tank leaks on the environment, (3) identify significant mineralogical properties of each stratigraphic unit for use in determining the mineral-waste reactions that affect the stability of radionuclides in the sediment column and, (4) assist Tank Farm Surveillance in determining the status of tanks and determining, on an as-needed basis, the spatial distribution of radionuclides leaked to the environment.

Modification of 13 of the 14 existing water wells in tank farms is complete except for capping. Approval for lockable water tight seals is still needed. The remaining well, located in 241-BY Tank Farm, is blocked by equipment and piping. Tank Farm Management is having the piping removed.

Five wells around Tank 241-A-104 have been deepened from 23 meters to 38 meters to determine if the leak plume had spread downward and laterally. One change in lithology was breached and no contamination was found.

Three 30-meter and one 60-meter deep starter wells were completed in preparation for the coring of additional wells to basalt. One hundred eighty meters of coring at 2 of 8 sites has been completed in 200 West Area. Core recovery is approximately 90 percent. Data from these wells, when combined with cable tool wells data, will provide sufficient geological information to determine the geology, hydrology, and mineralogy of the 200 Areas Control Zone.

Work is continuing on Tank 241-T-106 leak assessment. Work in progress includes: (1) analysis of tank farm surveillance data, (2) reduction of gamma energy analysis data, (3) completion of down well calibration tank, (4) drafting, (5) determination and analysis of moisture content of soils around Tank 241-T-106, (6) Pu analysis using copper foils in wells around tank and, (7) contouring radionuclides in plan and section view using computer programming.

Seismic Studies - K. R. Fecht

The objectives of this activity are: (1) to implement a system for detection of local earthquakes and a plan for evaluation of their impact on major waste containment systems, (2) to establish an inspection system for verifying the integrity of critical waste-management structures, and (3) to assess the regional tectonic stability with respect to current and long-term waste management plans.

The immediate response seismic monitoring array, to minimize releases of radionuclides to the environment due to earthquakes, has all four stations in operation. The stations are to be field tested and calibrated before going on line.

A progress report on the structure and stratigraphy of a thick sequence of Columbia River Basalt on the north flank of the Horse Heaven Hills from Wallula Gap eastward to Van Sycle Canyon was completed. The report includes bulk-rock chemistry and petrography of the basalt and breccias, geologic maps and cross sections of the study area, and structural and stratigraphic interpretations.

6.4 RISK ASSESSMENT

Hazards Available - L. E. Bruns

The objective of this activity is to devise mathematical relationships that equate toxic substances with "hazards available to man." Such relationships can be used to provide direction as well as to develop criteria for control of contamination sources. The effort should be helpful in predicting the Hazards Available (HA) in case of episodic events, such as range fires, high winds, explosions, and earthquakes.

A presentation on HA was given at the Waste Management 1976 Symposium in Tucson, Arizona, October 5, 1976. Much interest was generated and as a result contacts were made both in the United States and foreign countries concerning this type of approach. Dr. Newell Stannard of the University of Rochester, along with others, is preparing a book on "Relationship of Hazard and Toxicity for Plutonium and Other Substances," sponsored by Electric Power Research Institute. Atlantic Richfield Hanford Company has been asked to help review their report.

6.5 SPECIAL STUDIES

Gaseous Waste Monitoring Study-Battelle-Northwest

The objective of the BNW study is to determine the particle collection efficiency of selected ARHCO gaseous effluent monitoring systems and to provide recommendations on needed improvements.

The final report covering all experimental findings at the 291-B-1 (B-Plant stack) monitoring system was issued in November 1976. A theoretical evaluation of four gaseous effluent monitoring systems in the Tank Farms (296-A-12, 296-A-17, 296-P-3 and 296-S-18 stacks) has been initiated.

All sites have been reviewed by BNW and documentation of findings for the 296-A-17 gaseous effluent monitoring system was received. Findings on the three other systems being studied will be documented in the second quarter of FY 1977.

Experiments completed at the 291-Z-1 stack (Z-Plant stack) include two particle sizing experiments and all of the inert tracer experiments needed to evaluate the existing system. At least one more particle sizing experiment and completion of the analytical work on the tracer samples will be necessary before findings are documented. Tracer experiments have been completed at the 296-Z-3 stack (241-Z sump) and analysis of these samples is approximately 50 percent complete. At least one more particle sizing run will be completed before findings are documented. Detailed experimental findings for both of these systems will be documented in the second quarter of FY 1977.

Development of an economical tracer technique progressed considerably during this quarter. Experiments with titanium dioxide and zinc sulfide tracer materials have shown that comparison samples are feasible. Use of the tracer without a BNW comparison sample will be pursued in the second quarter of FY 1977.

7.0 ANALYTICAL METHODS TECHNOLOGY DEVELOPMENT

The objective of the analytical methods technology mission is to provide the integrated development of instrumentation and procedures to measure the concentration of radioactive and nonradioactive constituents in support of Production, Separation, Waste Management-Low-Level, and Research. Each activity, when completed, will provide an effective system by which Analytical Services can increase throughput of samples and/or better meet the needs of the operation being supported. Automated instrument control and automated data reduction is included when its use will improve the quality or quantity of information or will improve the safety aspects of an analysis.

7.1 ANALYTICAL INSTRUMENTATION DEVELOPMENT

The objectives of this program are to develop all phases of analytical instrumentation to (1) provide cost-effective and reliable support for ARHCO programs and, (2) reduce radiological exposure by requiring less 'hands on' attention by analytical personnel. Automation concepts will be evaluated to reduce human error, thereby improving efficiency and accuracy.

Subcritical Assembly for Low-Level Analysis

The status of the CFX system by IRT Corporation is summarized as follows:

Safety Analysis Review	- Complete except for publishing
Mechanical Design	- 82% Complete
Mechanical Fabrication	- 59% complete
Electrical Design	- 98% complete
Electrical Fabrication	- 59% complete

The CFX core tank was received and accepted. Design of the ^{252}Cf source transfer mechanism and construction of the demineralizer are in progress. Fabrication of the CFX appears to be about six weeks behind schedule. IRT Corporation SNM inventory limits have resulted in a delay in the receipt of the fuel plates until the third week of January. Construction of the CFX site at the 222-S Laboratory has begun and is scheduled for completion in April.

Fabrication drawings for the delayed neutron counter (DNC) grid structure were completed by IRT. Construction of the grid structure has not started. Electronic design for the DNC is 95 percent complete and fabrication is 50 percent complete. The BF_3 tubes for the system were received during this report period. Testing of the pneumatic terminus is 50 percent complete. Assembly and testing of the DNC is about six weeks behind schedule.

The Scorpio 3000 CFX control and data acquisition system was received and installed the first week of December 1976. Testing of interfaces and software is in progress. The receipt of this equipment was six months behind the original schedule. The impact of this delay cannot be evaluated until equipment and software evaluations are completed.

A document was issued as a guide for reviewing and preparing the more formal Safety Analysis Review. The first draft of the CFX transfer system exhaust control design has been completed and is being reviewed. Work will continue in off the site construction, Scorpio 3000 interfacing, ARHCO Safety Analysis Review, and the pneumatic transfer system in the next period. Design work on the exhaust control, cell interlocks, and rabbit handling systems will continue.

Radiation Analysis Systems Development - W. H. Zimmer

Ten Detector Array System. This project will be delayed by three to four months because of manpower shortages and difficulties in fabrication of the shield and relocation of laminar downflow hoods in the counting room.

Gamma, X-ray, Data Reduction Systems. Operation of NOVA 1220 computers in a sequential mode with the Real Time Disk Operating System (RDOS) and the stand alone gamma reduction (ARIS-II) program was successfully implemented. This reduces data reduction times by a factor of two.

The "D" detector used for backup to the primary reference detector was lost due to vacuum failure and will have to be replaced in the future.

Instrument operating procedures are being written for CTL gamma data acquisition-reduction equipment for use by Analytical Laboratory personnel. Training of Analytical Laboratory 222-S counting room personnel is scheduled for after the first of the year on the use of the ARIS-II gamma-ray data reduction system. Following this training, these personnel will assume the responsibility for analysis of all routine samples.

During the next report period, the Scorpio 3000 system will be received and tested. Interface requirements between Scorpio, which uses a PDP-11 and the existing NOVA 1220 will be completed.

Alpha Counting Instrumentation - J. K. Hartwell

An automated multi-input alpha energy analysis (AEA) system designed for data acquisition on eight samples simultaneously has been tested with four detectors in place and found to function well. Data reduction is still performed manually; however, an initial test of an AEA software package has been completed. This package, running on the CYBER computer, receives raw spectral data via the NOVA 840 data link. The software system correctly identified the isotopes in 13 test spectra. The test spectra included data for two National Bureau of Standards plutonium isotopic standards. The CYBER computed spectral analyses were within five percent of the standard values. Data reduction of the remaining 11 spectra collected from process samples qualitatively identified the spectra correctly.

Work is continuing to complete the remaining vacuum-detector systems, to complete testing the spectral data, and to document the system. The transfer to routine operation is expected to be accomplished on schedule.

Beta Counting Systems - J. K. Hartwell

Parameters for measuring iodine-129 and other low energy beta emitting isotopes are on the milestone schedule.

This program in support of Waste Management, Purex, and effluent analyses is designed to identify optimum separations techniques and instrumentation to measure the concentration of the isotope of interest.

An improved "separation still" has been implemented which has resulted in recoveries of 99.27 percent $\pm$ 2.86 percent of the accepted standards values. Decontamination factors exceeding 1×10^4 have been demonstrated on synthetic solutions of typical fission product mixtures.

Screening tests have been completed using a factorial design experiment to evaluate the detection instrument operating parameters, scintillating media, quenching characteristics, and sample-to-scintillator volume ratios. A scintillation mixture (cocktail) containing 8 g/liter 2-(4'-t-butylphenyl)-5-(4'-biphenyl)-1,3,4 oxidiazole - 0.5 g/liter 2-(4' biphenyl)-6-phenyl-benzoxayole in toluene has been identified superior and will be tested further.

Decolorization studies of the sample solutions will be continued during the next report period. Statistics will also be accumulated on standard mounts during the next period.

Inductively Coupled Plasma Emission Spectrometer - C. S. Homi

Misunderstandings of ARHCO's requirements for the system by Applied Research Laboratories have delayed design of the plasma containment box. The spectrometer is not expected to be received until February 1977. Standards preparation for the equipment was initiated and quartz stills were received for the preparation of high purity H₂O and acids.

Successful meeting of milestones will depend heavily on the date of receipt for the equipment and the time required to construct and install the scrubber and off gas systems.

7.2 ANALYTICAL METHODS DEVELOPMENT

Analytical methods and technology are developed within the scope of this program. Specific analytical methods are required to:

- o Measure organic degradation products in processing streams.
- o Assay Purex input, product and waste streams.
- o Improve environmental sampling and analysis techniques.
- o Improve measurement techniques for Waste Management activities.
- o Improve measurement methods for plutonium production.
- o Provide short term analytical assistance.

Titrimetric Uranium Assay - J. R. Jewett

The Davies and Gray titrimetric method for determination of uranium was studied using both potentiometric and amperometric end point detection. Potentiometric end points were unstable but amperometry gave a very sharp endpoint. Average deviations of 0.06 percent were obtained by amperometry; however, results had a negative bias of 0.27 percent.

Future efforts will be directed at removing the bias, scaling down the method to smaller volumes, and evaluating a constant-current coulometric method for U which is based on the Davies and Gray Method.

Actinide Analytical Program - S. J. Johnson

Evaluation of electroplating techniques and parameters has been impaired by the unavailability of platinum electrodes. The electrodes have now been received and the project should return to the milestone schedule.

Separation and measurement parameters are being identified through Plackett-Burman experiments to determine which of six variables most affect the Aliquat-336 - dibutyl-N, N-diethylcarbamylmethylene phosphonate (DBDECMP) system. The nitric acid wash step and a bisulfate fusion before electroplating appear to enhance plutonium and neptunium recoveries. Europium extracts with thorium and causes high results due to the europium-arsenazo complex which is formed and is measured spectrophotometrically with the thorium-arsenazo complex.

New purification techniques have been defined for dihexyl-N,N-diethylcarbamylmethylene phosphonate (DHDECMP), a similar extractant to DBDECMP, which now makes this an available reagent for extracting the actinide constituents of waste. The advantages of DHDECMP will be exploited during the next report period.

Analytical Development for Waste Management

Low Energy Beta Isotope Measurement - J. K. Hartwell

As reported in beta counting systems, above, the low energy beta isotope measurement program is on schedule. Included in this program is the separation and measurement of iodine-129, samarium-151, promethium-147, carbon-14, and technetium-99.

Gas-Liquid Chromatography - S. J. Johnson

A new gas chromatograph has been received for development of parameters to quantify organic degradation products from process streams accumulated wastes and reagent purity. This project is slightly behind schedule on receipt of the

equipment; however, unless significant deviations are found in instrument operation, testing should be completed during the next report period as scheduled.

Carbon Rod Atomic Absorption - T. J. Hanson

The activity to optimize and implement the carbon rod atomizer on the atomic absorption spectrometer has been proceeding on milestone schedule.

This unit, a flameless atomizer, appears to provide the advantages envisioned, i.e., one to three orders of magnitude greater sensitivity, and a much smaller sample required for an analysis. Tests to date indicate precision and accuracy remain about the same for flame and flameless systems. Advantages to be gained then are: (1) radiological exposure to employee's can be reduced both while making dilutions and while making the measurements and, (2) detection limits of the elements can be lowered without significant loss of precision or accuracy, thereby providing the basis for better process control.

The technology developed to date will be transferred to the analytical laboratory for continued work until a replacement can be provided for additional applications development.

Mercury in Waste Tank Samples - T. J. Hanson, A. O. King

A sensitive precise measurement system has been developed to provide mercury concentration measurements in waste tank contents. Parameter optimization has been completed on standardized material and is ready for tests using radioactive samples as soon as contamination control equipment is in place. Detection level for the method is 0.001 μg Hg with a standard deviation of less than 1.0 percent at the

95 percent confidence level, and accuracy of 100 percent on standard solutions. Interferences from 27 ions were investigated and found to be insignificant.

Additional work on this activity will also be slowed by the termination of the principle investigator.

Standards Preparation and Measurements - A. G. Miller

Measurement of physical properties of waste tank solids is being investigated. This activity, leading to the development of physical standards, is being initiated ahead of the milestone projection to fulfill the need of the Salt Cake Characterization program, now in progress. The immediate need is a measurement method for particle size distribution of water soluble crystals. Methods are required to remove interstitial liquor and distinguish agglomerates from individual crystals.

During the next report period, a microscopic evaluation of the effect that proposed screening, agglomerate reduction, interstitial liquor removal, and organic suspension media have on the particle size distribution of synthetic salt cake, will be conducted.

Analytical Assistance

X-Ray Spectrometer - S. G. Metcalf

An X-ray fluorescence spectrometer which has not been fully utilized is being repaired and developed for service to the Analytical Laboratory. This unit will extend the quantitative identification of elements present in test samples. Time savings will be realized by implementing the automatic sample changer which is an integral part of this instrument. Optimization of operating mode parameters is in progress.

Sample preparation techniques such as the Claissé fluxer are proceeding ahead of the milestone schedule. The need for definitive procedures have developed more rapidly than projected. A literature search was completed for X-ray sample preparation by fusion techniques.

Beta Counting - J. K. Hartwell

The beta emission analyzers of the Analytical Laboratories are being modified to provide higher throughput. The instruments have dual detectors, one manually and one automatically operated. The modifications when complete will allow automatic simultaneous operation of both detectors.

Electronic compatibility problems have been resolved. Implementation and optimization is expected to be performed by Analytical Laboratory chemists.

Sludge Analyses - Z Plant - S. J. Johnson

Gas chromatographic analyses were performed on aqueous materials, organics, and solids from Waste Treatment, 242-Z. After appropriate washes eight unidentified constituents were separated in addition to TBP, DBBP and CCl_4 which are normally expected to be present.

7.3 ANALYTICAL COMPUTER SYSTEMS - D. L. Anderson

The objective of this activity is to interface all major analytical instrumentation to a computer system for control, data acquisition and data reduction.

A program was developed to process air sample data from the 222-S Laboratory and produce a daily report for monitoring purposes. The daily report can be completed four to five times faster (less manpower) than the routine CYBER

link. However, the 222-S computer system will not be used routinely until a high speed data link to the CYBER is established in the near future. The air sampling system will be further enhanced when direct links between the 222-222-S computers and air sample counters are established.

The hardware to perform instrument control and data acquisition from the Spex Raman spectrometer has been assembled and tested. Initial software to automatically scan a wavelength region of interest and store the spectrum in the computer has been written and tested. This brings both computers to the maximum memory available for programs. Any additional memory would only improve the execution speed and is not necessary at this time.

The Analytical Laboratory gamma counting program has been rewritten by R. L. Kubischta, Laboratory Management and Science Systems Section, so that it may use only one of the two computers in the 222-S system. This permits one computer to be used for development work on day shift without interrupting counting room service. Also, if one computer is lost due to hardware malfunction, backup service can be provided with the other computer with nearly full capability.

A new combination paper tape reader/punch was received and installed. This will permit storage of programs on an independent medium to back up disc integrity. Memory in both the NOVA 830 and NOVA 840, 222-S Laboratory computers, has been increased to 96K words. This brings both computers to the maximum memory available for programs. Any additional memory would only improve the execution speed and is not necessary at this time.

7.4 FIELD INSTRUMENT DEVELOPMENT

The activity objective is the development of field instrumentation which can be used to obtain data on the physical and radiochemical characteristics today and changes with time of the various waste sites in the 200 Areas. It is also expected that this data, along with the instrumentation, will provide a basis for the long-term surveillance of these sites.

Beta Assayer - L. E. Bruns

The Beta Assayer for field assaying of strontium-90 and other beta emitters is completed and undergoing laboratory checks and calibrations. Some minor problems were experienced with the more sophisticated electronics, but "bugs" are being worked out and the unit should be ready in January 1977 for initial field tests.

Dev-Van-I - L. E. Bruns

The Dev-Van-I will house detectors and electronics that will allow surface assays of all types of facilities and ground surfaces for radionuclide concentration. All parts were received and the van assembly was initiated in December. The unit will be ready for field tests in the second quarter of FY 1977. Work on improved units for Dev-Van-I continued; a purchase requisition was sent to National Nuclear Corporation to test a silicon detector system initially in their laboratory. Part of the test will involve field testing surfaces in the 200 Areas plateau and assaying in-well samples that have been packaged for field measurement.

This unit should assay plutonium in the one nanocurie range. Should the unit prove valuable for *in situ* assaying, it could be purchased and kept by ARHCO for routine field work.

Work was completed on the intrinsic Germanium Pu-Am assayer. The system can measure the 60-Kev ^{241}Am and the combined L-X-rays from ^{239}Pu and ^{241}Am . From this the concentration of ^{241}Am can be determined and the americium contribution subtracted from the L-X-ray reading to give ^{239}Pu . Depth corrections can also be made by ratioing the ^{241}Am 26- and 60-Kev peaks. A letter report written by Battelle Northwest is available entitled, "Development of a Plutonium-Americium Monitor for In Situ Soil Surface and Pond Bottom Assay." This instrument has similar sensitivity to the silicon detector.

Dev-Van-II - L. E. Bruns

Proposals were received for a van-logger and only one met deadlines and specifications, Gearhart-Owen Industries, Inc. The van will have suitable equipment to allow down-hole assay of radionuclides to accurate and precise depths in the pico to nanocurie range. A Government Services Administration waiver was obtained, and agreement was reached on what portion of the cost was capital and what portion was expense.

A proposal was received from Humphrey, Inc. on a directional unit for down-hole logging. Some development will be required on the final direction unit for down-hole logging. The final directional gyroprobe, sensor positioning device will cost about \$12,250 including development costs. The cost of the control box to be mounted in the van will cost

\$5,500. This unit will be able to give the direction, within five degrees, that the window of the radionuclide detector is pointing.

A report on work sponsored by ARHCO was completed by Battelle Northwest. It is entitled, "In Situ Subterranean Gamma-Ray Spectroscopy," BNWL-SA-5912 by H. L. Nielson, N. A. Wogman and R. L. Brodzinski. It discusses theory, laboratory, and field work concerning use of down-hole gamma detectors like Ge(Li), intrinsic Ge and NaI(Tl) units.

"Passive" Activation System - L. E. Bruns

Use of metal foils sensitive only to neutrons can be valuable for estimating the plutonium concentration in the subsurface sediments, on surfaces, and in hoods, pipes, etc. A small calibration facility in the 300 Area was completed and one source, 100 g of PuF_4 , was sent to Battelle Northwest for initial studies. Moisture readings were taken on the sand in the calibration tank as obtained from the field and the moisture was relatively constant throughout the tank, 4-5 percent. Approval from U. S. Energy Research and Development Administration (ERDA) has been obtained for formulating three PuO_2 sources; Am sources are available. Tests in the 300 Area will help verify type of foils, optimum foil geometry, and initial moderation and attenuation effects. Final calibration for field use will take place in the 200 West Area Calibration Facility.

To assist the Tank 106-T Study, metal foils were placed in 106-T monitoring wells. All initial depths were at 12 meters (same as bottom of tank); values ranged from background to 1340 nanograms Pu/cm^3 of soil. Varying depths will need to be tried to ascertain if it is Pu in the sediments or whether it is a neutron response from the tank.

Calibration Facility - L. E. Bruns

The fabrication and emplacement of all three calibration tanks were completed. This includes the large 3 meter diameter point source and the two 1.2 meter diameter 4π tanks. All matrix materials for the large diameter vessel have been prepared or are readily available; these include burial ground type waste, salt cake, clean sand (control), and "average" back-fill materials. The Calibration Facility is just south of the Redox 202-S Building.

Void Volume Detector - L. E. Bruns

A proposal was received from Stanford Research Institute for detecting void volumes in the subsurface. It is similar to the one proposed by Holosonics (see last Quarterly Report).

New Instrumentation - L. E. Bruns

Discussions with Princeton Gamma-Tech, Scientific Applications, Stanford Research Institute, and Intelcom Radiation Technology resulted in requesting proposals from these companies on various instrumentation. The major one, at present, is the development of a monitor for assaying density, porosity, void volume, and moisture of high-level waste salt cake. The main approach to date is use of a ^{252}Cf activation system. Other possibilities include ultrasonics.

8.0 ENVIRONMENTAL CONTROL TECHNOLOGY

8.1 MANAGEMENT OF SURPLUS FACILITIES AND EQUIPMENT

Disposition Methods Research and Development

Contaminated Equipment Volume Reduction - G. A. Beitel (D. C. Nelson, Program Manager)

This program proposes to demonstrate that multi-ton, stainless steel vessels can be cut into small pieces and subsequently melted, thereby reducing the storage volume by up to a factor of 20, in addition to greatly improving the quality of the material to be stored. Emphasis is being placed on the arc saw, the size-reduction tool, and a cold-wall vacuum furnace for melting. Cold testing of size-reduction and volume-reduction equipment will begin by the end of FY 1978. Hot testing of size-reduction equipment will begin in FY 1981, and volume-reduction testing with contaminated materials will begin in FY 1982.

The design of a test stand on which the arc saw will be tested was begun and will be completed under this program. The construction, installation, and testing of the saw will then be supported by another program, "Decontamination and Decommissioning of Nuclear Fuel Cycle Equipment and Facilities." This will enable the CEVR Program to devote all its efforts to the meltdown demonstration.

One task, the decontamination of a Purex dissolver, was not completed last year primarily because of the strike, and has been carried over to the current year. No progress was made this quarter, however, because higher priority work preempted work planned for the decontamination section.

Detailed design of the cold-wall vacuum furnace completed during the prior quarter was reviewed and several minor corrections were made. Preparation of the preprocurement report and purchase requisitions for the vacuum furnace were begun and are about 60 percent complete. A cost estimate for the fabrication and installation of the complete furnace system was begun and is about 50 percent complete.

The facility in which to install and test the furnace has not been selected. A preferred option, the construction of a special purpose building does not appear feasible because there are no General Plant Project funds available. (It is estimated that such a building would cost about \$220,000.) Three other sites are currently being considered: (1) the old U Plant mockup facility which is ideal with respect to size, location, and utilities; (2) a retired plutonium processing building which will require extensive decontamination and modification that could easily cost more than a special purpose building; and (3) the old test reactor building which is of the proper size and could be used if the decontamination process, begun in September but suspended, was completed at a cost of about \$120,000.

A new version of the arc saw test stand was designed. The new design strongly resembles a vertical rail mill. The saw will be able to cut in both the vertical and horizontal planes and will dismantle a test object by cutting from one side and with a sequence of vertical, then horizontal dicing cuts. The test object will be dismantled in one-foot thick layers (as in cliffside quarrying) in contrast to our earlier concept which proposed to dismantle the object from top-to-bottom in horizontal planes (as in open pit mining). A high volume water flood will keep the blade underwater, but neither the saw nor the workpiece will be submerged. Title I

design was completed and a cost estimate of \$75,000 (hardware fabrication only) was provided. Title II design is about 50 percent complete.

Purchase requisitions were prepared for the high current power supply for the furnace, and a small (1 kilogram) induction furnace, which will be used for contaminated metal melt tests.

REFERENCES

1. ARH-CD-149B, (unclassified), November 1976, "ARHCO Quarterly Report, Process Technology and Process Development, July 1976 Through September 1976," pg. 50.
2. Ibid, pg. 52
3. Ibid, pg. 51

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