ATLANTIC RICHFIELD HANFORD COMPANY
QUARTERLY REPORT
PROCESS TECHNOLOGY AND PROCESS DEVELOPMENT
APRIL 1977 THROUGH JUNE 1977

Research and Engineering Division

August 1977

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Energy Research and Development Administration nor any of their employees, nor any of their contractors, subcontractors or their employees makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

Atlantic Richfield Hanford Company
Richland, Washington 99352

geg

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

TABLE OF CONTENTS

	<u>Page</u>
ABSTRACT	vi
INTRODUCTION	1
1.0 <u>WASTE MANAGEMENT (COMMERCIAL)</u>	
1.1 NATIONAL WASTE TERMINAL STORAGE PROGRAM . .	2
Basalt Feasibility Studies	3
Waste Containerization Studies	5
1.2 WASTE PROCESSING-COMMERCIAL, RESEARCH AND DEVELOPMENT	7
Remote Maintainability of Transuranic Waste Processing Equipment	7
1.3 SUPPORTING STUDIES-RESEARCH AND DEVELOPMENT	7
Sealed Storage Cask Demonstration . . .	7
Decontamination and Decommissioning Technology for Nuclear Fuel Cycle Equipment and Facilities	10
Special Studies	12
Generic Environmental Impact Statement	12
2.0 <u>SEPARATIONS, PRODUCTION, AND SCRAP RECOVERY</u>	
2.1 SEPARATIONS-PUREX	13
2.2 PRODUCTION-Z PLANT	15
2.3 SCRAP RECOVERY-Z PLANT	18

TABLE OF CONTENTS (Continued)

	<u>Page</u>
3.0 <u>CURRENT WASTE MANAGEMENT (DEFENSE)</u>	
3.1 WASTE FRACTIONIZATION	23
Strontium Recovery	23
Cesium Recovery	23
3.2 ENCAPSULATION	25
Tilt-Pour Cesium Equipment	25
Alternate Compounds for Encapsulation	26
Hot Pressing Strontium Fluoride	26
3.3 WASTE CONCENTRATION	27
Waste Concentration Process	
Technology	27
Residual Liquor Disposition	29
Salt Cake and Sludge	
Characterization	30
3.4 TANK FARM SURVEILLANCE AND MAINTENANCE..	31
4.0 <u>LONG-TERM MANAGEMENT OF DEFENSE WASTE AT HANFORD</u>	
4.1 STORAGE SYSTEM INTEGRITY AND ENGINEERED IMPROVEMENTS	35
Storage System Integrity	35
Engineered Improvements	45
4.2 WASTE RETRIEVAL	45
Conceptual Design-Prototype	
Retrieval System	46
Tool and Equipment Development	46
Engineering Studies and Analyses	47

TABLE OF CONTENTS (Continued)

	<u>Page</u>
4.3 WASTE PROCESSING	48
Immobilization Processes - Laboratory Development	48
Immobilization Processes - Equipment Development	50
Radionuclide Removal Process	57
4.4 STORAGE AND DISPOSAL ALTERNATIVES	69
4.5 ALTERNATIVE STUDIES AND ASSESSMENTS	70
Analysis of Alternatives	70
Hanford Defense High-Level Waste Document	72
Programmatic Environmental Impact Statement	73
5.0 <u>LONG-TERM MANAGEMENT OF CONTAMINATED SOILS AND SEDIMENTS</u>	
5.1 CONTAMINATED SEDIMENTS CONTROL	76
Site Characterization	76
Trench Characterization	77
Sediment Stabilization	79
6.0 <u>ENVIRONMENTAL IMPACT TECHNOLOGY</u>	
6.1 HYDROLOGY	82
Groundwater Hydrology	82
Offsite Migration	84
Radionuclide Migration in Groundwater	86
Long-Term Groundwater Management	87
6.2 TRANSPORT	89
Moisture Transport Studies	89
Biotic Transport	91
6.3 SEISMOLOGY AND GEOLOGY	93
Tank Farm Geology	93
Seismic Investigations	97

TABLE OF CONTENTS (Continued)

	<u>Page</u>
6.4 RISK ASSESSMENT	98
Hazards Available	98
6.5 SPECIAL STUDIES	99
Gaseous Waste Monitoring Study, Battelle-Northwest	99
7.0 <u>ANALYTICAL METHODS TECHNOLOGY DEVELOPMENT</u>	
7.1 ANALYTICAL INSTRUMENTATION DEVELOPMENT . .	100
7.2 ANALYTICAL METHODS DEVELOPMENT	109
7.3 ANALYTICAL AUTOMATION	113
7.4 FIELD INSTRUMENT DEVELOPMENT	116
8.0 <u>ENVIRONMENTAL CONTROL TECHNOLOGY</u>	
8.1 MANAGEMENT OF SURPLUS FACILITIES AND EQUIPMENT	122
Disposition Methods Research and Development	122
Contaminated Equipment Volume Reduction	122
REFERENCES	123
DISTRIBUTION	124

ABSTRACT

This quarterly report is the fifth 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 April through June 1977.

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 represents 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 Waste Management Production and Reprocessing (WPR).

Previous periodical reports covering segments of this report's subject material were published as ARH-ST-110 A through D, ARH-ST-128 A and B, ARH-ST-132 A through C, ARH-ST-139, ARH-LD-111 A through E, ARH-LD-149 A and B, ARH-LD-152 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 commercial radioactive waste. An Office of Waste Isolation (OWI) has now been 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. The 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.

Basalt Feasibility Studies - R. A. Deju

(R. E. Smith - Study Manager)

The objective of this study is to obtain information about basalt to allow a full assessment of the feasibility of using basalt as a terminal storage medium for commercial radioactive waste, while assuring environmental safety, permanent isolation, and retrievability during the early confirmation and demonstration period.

During the reporting period April 1, 1977 to June 30, 1977, work progressed in drilling a new core well (ARH-DC-2). This core well is ultimately expected to penetrate the Umtanum basalt flow at a depth of about 3,000 feet below land surface. By the end of the quarter, a core rig had been installed onsite and drilling to a depth of 1,250 feet had been completed.

Well RSH-1 atop Rattlesnake Hill has been cleaned out, logged, and tested. Results of tests conducted in that well have been interpreted. Permeabilities identified in the various strata are in the nanodarcy range with slight permeabilities in the rubble zones. Additional drill stem tests are scheduled for next month and, after present completion of these tests, piezometers will be installed to isolate various horizons.

Geologic studies at Sentinel Gap, Umtanum Ridge, and Rattlesnake Hills are proceeding on schedule. Field reconnaissance of all three areas is nearly complete. Samples have been taken at various locations and are being analyzed in the laboratory for paleomagnetic and geochemical properties.

The waste-form basalt compatibility schedules are going on schedule.

Short-term experiments are being carried out on the pairs: basalt (Umtanum, Umtanum-like pillow basalt, BCR-1 standard basalt) and waste form (Barnwell-type PW-7a calcine, PNL Borosilicate glass 76-68, supercalcine SPC-2 and simulated

spent LWR fuel) in order to evaluate overall stability. Preliminary results for a dry repository at near atmospheric pressure suggest no gross interactions at 200° C. Detailed analyses are being carried out on these, as well as similar runs at 600° C.

Compatibility experiments are also underway simulating conditions representing a collapsed dry repository at ambient pressure (400 bars and 400° C) and a flooded repository at ambient pressure and 200° C.

The plans to study all the environmental factors needed to establish the feasibility of storing radioactive waste in Columbia River Basalts have been compiled in a draft document which is presently being reviewed by our staff. A parallel document to define geoengineering studies to be conducted is being prepared.

An additional well, ARH-DC-3, will be drilled during the coming quarter. Plans for this well and procurement of services for the drilling have been completed during the present quarter.

The staff visited the Bacon Siphon and Tunnel No. 2, a railroad tunnel near Kahlotus, Washington, and a tailrace at Snoqualmie Falls Power House. Photographs were taken of all these facilities. Of particular interest is the tailrace tunnel under Snoqualmie Falls, which is a dry facility, 12 feet wide, 27 feet high, and 450 feet long. It has been in service since 1898. Photographs of all these facilities were transmitted to the Office of Waste Isolation during the month of May.

All milestones are presently being met.

Waste Containerization Studies - D. A. Turner,
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 and spent fuel elements. The prime objective of Task II is to develop design criteria, specifications, and designs for waste containers for all radioactive waste except high- and low-level wastes and spent fuel elements.

Task I - High-Level Waste Canister Envelope Study -
E. L. Moore

Comments were received from the Office of Waste Isolation (OWI) on the draft of the Preliminary Canister Size Criteria document. These were incorporated and the document was issued as ARH-ST-151. The information provided in the document is needed for the preliminary design studies of the pilot plant repositories by OWI, which are currently in progress.

Contacts were maintained with organizations who are actively participating in various activities which comprise the canister cycle to insure that the canister design criteria are compatible with existing and planned fuel cycle facilities. A meeting was held in Richland on May 12, 1977 between Atlantic Richfield Hanford Company and OWI representatives to review the current program status.

Contract procurement for obtaining technical support services to provide data for the canister design criteria was continued. Five contracts are involved and three have been

placed. These were the preliminary engineering study of shipping casks to Science Applications, Inc., the canister materials and fabrication study to Battelle Columbus Laboratories, and the canister structural analysis to Kaiser Engineers. Meetings were held with the first two firms to initiate the studies. A meeting with Kaiser Engineers is planned. Placement of the remaining two contracts is crucial to the completion of the initial study phase of the program by September 30, 1977.

A remotely operated canister handling device is being developed by ARHCO. A plastic working scale model was constructed and operated. A full-size device is being fabricated from metal for cyclic testing, followed by destructive examination. The device latches on to a bail attached at the top of the canister. The attractive feature of the design is that the device is self-centering and self-orienting with respect to the position of the hook and canister bail.

Task II - Radioactive Waste Packaging Study - D. B. Calmus

The draft plan for the Radioactive Waste Packaging Study, issued in March 1977, requested the development and issue of a Preliminary Status Report on July 29, 1977. A draft of the Preliminary Status Report has been completed and is now being prepared for comments. This status report will include a tentative scope of waste container criteria, accomplishments to date, and potential problem areas.

Contract procurement action for obtaining technical consultation services is now underway. Plans at this time are to add on to the proposed five contracts for the Task I - High-Level Waste Canister Study. Two contractors were visited in June to discuss "adding on" work in the areas of corrosion, (Battelle Columbus Laboratories), and transportation, (Science Applications, Inc.).

On May 10, 1977, Mound Laboratory representatives, Mr. J. B. Peterson and Mr. J. M. Garner, visited Hanford to observe burial practices and discuss waste forms and packaging. On May 12, 1977, representatives for the Office of Waste Isolation, Mr. J. M. Morrison, Mr. R. S. Lowrie, and Mr. W. H. Pechin, visited Hanford for a study plan review.

1.2 WASTE PROCESSING - COMMERCIAL, RESEARCH AND DEVELOPMENT
Remote Maintainability of Transuranic Waste Processing
Equipment - K. H. Henry (D. A. Turner, Program Manager)

The objective of this program is to provide practical solutions to selected problem areas in the remote operation and maintenance of transuranic waste processing equipment in nuclear fuel cycle plants. These solutions will be provided by establishing design bases through engineering studies, conceptual design, and hardware development and demonstration. Transuranic solid waste incineration equipment has been chosen to receive initial emphasis under this program.

Progress on this program was interrupted early in the report period due to staffing requirements for the Spent Unreprocessed Fuel Facility Program. Personnel reassignments are planned which will allow resumption of this program.

1.3 SUPPORTING STUDIES - RESEARCH AND DEVELOPMENT

Sealed Storage Cask Demonstration - D. A. Turner, Program Manager

The Federal government has agreed 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 being developed under the

National Waste Terminal Storage Program. In the event that these final storage facilities are not available when high-level radioactive waste deliveries begin, above ground storage in a Spent Unreprocessed Fuel (SURF) Facility 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. Design verification of the SSCC will be completed during fiscal year 1977.

Conceptual Design Support Studies - J. M. Davis

An SSCC demonstration unit has been built and activities are being conducted to evaluate the integrity and feasibility of the design. 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 SURFF, (4) evaluation of canister and cask handling methods, and (5) fabrication of an SSC to be loaded with radioactive waste or spent fuel elements to demonstrate the feasibility of the concept. Phases 1 through 5 are currently in progress.

Heat Transfer Design Verification. The collection of temperature data from the electrically-heated demonstration unit is now complete. The unit has been disassembled and photographed. Operating the unit at 10 kilowatts with all vents plugged with concrete blocks resulted in cracking the exterior surface extensively above the midplane. Upon cooldown, the exterior cracks closed and the interior cracks remained. The cracking was expected and followed predictions that they were too small to cause a radiation shielding problem. Battelle-Northwest will complete its evaluation of the temperature and strain gage data in July 1977.

Structural Evaluation. The structural evaluation of the candidate welds is continuing.

Decontamination Evaluation. Battelle-Northwest has continued testing of contaminated canisters in the mini wet basin equipment. No progress has been made in decontaminating small test canisters using high-pressure water.

Demonstration of Remote Welding and Nondestructive Examination. All components of the demonstration were completed and installed in the test area, and the actual demonstration will be completed in July 1977.

Evaluation of Canister and Cask Handling. Design of the prototype container handling mechanism has been completed, and two units are being fabricated for extensive testing. The invention report is completed and patent possibilities are being investigated.

Demonstrations of Caisson Storage Mode for SURF. Three demonstrations of the caisson storage mode for Spent Unreprocessed Fuel (SURF) will be made. The surrounding soil temperature profiles will be measured, and the behavior

of various heat transfer media surrounding the heater will be studied. The site has been chosen; test holes have been bored for soil analysis; designs for caissons are in progress at Kaiser Engineers; and a small building for controls and instrumentation has been ordered.

Decontamination and Decommissioning Technology for Nuclear Fuel Cycle Equipment and Facilities - R. E. Smith, Program Manager

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 light water reactor (LWR) fuel cycle.

Size Reduction - G. A. Beitel

The objective of this program is to develop and demonstrate high-speed, remotely operated cutting techniques for dismantling radioactively contaminated equipment and facilities.

The arc saw is being adapted and further developed for application to the problem of dismantling stainless steel vessels up to three meters in diameter.

The contract was signed with ReTech, Inc. to design and build one complete arc saw system equipped with a nominal 40-cm diameter (50-cm maximum) blade, and a second saw head unit capable of accepting a nominal 120-cm diameter blade. The contract was begun on May 6, 1977 and delivery date is September 26, 1977. The preliminary drawings have been completed and approved. Approximately 30 percent of all components are on order and some parts are already being fabricated and assembled.

Design of the enclosure and the subsystems supporting the large arc saw was begun and is in progress.

Recent experimental data relating to the arc saw power supply operation was reviewed and used to rewrite the specifications for the power unit we are in the process of purchasing. This revision requires that the bids be rebid and will cause an additional delay in the receipt of the power supply. As a result, it is now probable that the delivery date may exceed December 31, 1977. The earliest date for operating the small arc saw is projected to be February 1, 1978; this is well ahead of program milestones.

A paper which is being submitted for presentation at the November 1977 American Institute of Chemical Engineers' meeting was written this quarter and is being routed for approval.

Decommissioning - L. E. Kusler

The objective is to review current decommissioning activities and to identify design requirements to permit acceptable decommissioning of LWR fuel reprocessing and recycle facilities at the end of their operating life.

Techniques used to decommission government and commercial facilities have been reviewed and a document summarizing results has been issued. One of the more promising techniques is the use of shaped charges to simplify cutting and material scarfing operations. If provisions are made in the design of new facilities for using this technique, decommissioning operations can be further simplified. Plans are to develop a specific detail design to fully evaluate this concept.

Characterization of existing chemical reprocessing plants has been started to help identify specific design features that will simplify decommissioning.

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 a Retrievable Surface Storage Facility for Interim Storage.

Atlantic Richfield Hanford Company efforts continued at a low level in support of the "Generic Environmental Impact Statement for Managing Radioactive Wastes from Commercial Reactors and Post-Fission Operations in the LWR Fuel Cycle (GEIS)." Document preparation is being coordinated by BNW. Atlantic Richfield Hanford Company participation during the quarter was limited to responding to specific questions from BNW. Little, if any, further work in support of this document is foreseen.

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, development of automatic control systems for the vacuum fractionator and process condensate recycle system, improvements in high-level waste denitration, and improved operating methods for acid absorbed are to be completed prior to plant startup. Methods for controlling NO_x and ^{129}I emissions will be developed for possible installation during plant operation.

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 changes in the plant which have been adopted since initial startup. About 15 percent of the rough draft has been prepared and some drawings have been started.

NO_x Control/Vacuum Fractionator Control - S. S. Koegler

Assembly of the pilot plant bubble cap tower has begun. Stainless steel tubing required for the project is not available and has been back-ordered. All column components for the NO_x scrubber have been received except for instrumentation (including the NO_x monitor). Additional components for the vacuum fractionator are on order.

Scouting studies are being conducted by the Chemical Laboratory to identify an effective scrubbing media for final NO_x cleanup. Sulfuric acid is being considered because it can be easily regenerated.

Denitration - J. W. Gehrke

Because waste denitration has limited the overall Purex Plant production rate, the current sugar denitration process and alternate processes will be investigated to improve the denitration time cycle. Laboratory tests will study and compare optimization of sugar denitration and potential replacement processes in the Zirflex flowsheet. Laboratory studies were initiated during June, 1977.

Control of Iodine-129 in Purex - D. W. Jeppson, D. M. Strachan

Methods are being determined to control ^{129}I releases from Purex during fuel reprocessing to "as low as reasonably achievable." The current plant silver reactors were determined to be poisoned with chloride and to contain essentially no iodine. Analysis of Purex chemicals for chloride concentrations indicate that chloride additions to the process via chemical additions are sufficiently low to allow a reasonable silver reactor lifetime in collecting iodine during fuel reprocessing. Chloride analysis of uranium metal is being performed to determine the potential chloride contribution to the silver reactors from the fuel.

During the last quarter, the ^{131}I tracer work was completed in the Chemical Technology Laboratory. Attempts to determine the extent of iodine volatilization from the dissolver during fuel dissolution were frustrated by inconsistent results.

Iodine volatilities ranged from 30 to 90 percent. A 99.8 percent volatilization of the tracer ^{131}I was achieved, however, by continuously adding a 30 percent solution of H_2O_2 and intermittent additions of $2.25 \times 10^{-3}\text{M}$ solution of KI to a simulated dissolver solution. Use of this process in the Purex Plant would reduce the operating lifetime of the silver reactors. Use of just H_2O_2 yielded a 90 percent volatilization of the tracer ^{131}I .

Since chemistry of the dissolution of irradiated fuel is much different than of nonirradiated uranium, plans for the next quarter include experiments with irradiated fuel from N Reactor which has been cooled less than 30 days. Completion of this aspect of the work is expected this quarter. Documentation of this work will be completed during the final quarter of this calendar year.

2.2 PRODUCTION - Z PLANT

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

Solvent Extraction Model - M. J. Klem, K. H. Oma

The purpose of this activity is to develop a mathematical model of the Plutonium Reclamation and Waste Treatment solvent extraction system using the SEPHIS computer program.

Plutonium Reclamation

Demonstration of the standard carbon tetrachloride (CCl_4) diluent flowsheet and the computer-derived tetrachlorethylene (TCE) and trichlorobenzene (TCB) alternate diluent flowsheets has begun using synthetic feed and laboratory mixer-settler

solvent extraction equipment. After allowance for the inefficiency of actual mixer-settler stages, agreement between SEPHIS-calculated and experimental results was excellent for the CCl_4 and TCE diluents. Testing of the TCB diluent flowsheet has not been started because of other priorities. In addition to flowsheet testing, work has begun on identifying the alternate diluent chemical and radiolytic degradation products.

The startup plans for the Plutonium Reclamation Facility (PRF) call for the dissolution of solids from the access glove box and canyon floor and then solvent extraction recovery of the plutonium. Laboratory analysis of the dissolved floor solids show the presence of 0.03 - 0.54M fluoride and 0.004 - 0.018M phosphate. The plutonium(IV) ion forms stable complexes with both these anions thereby reducing the extraction of plutonium by tributylphosphate. The SEPHIS computer program was modified for the retarding effects of fluoride and phosphate ions. With this change, the computer program can be used to establish limits in the solvent extraction feed.

Waste Treatment

The SEPHIS program was modified for the extraction of plutonium and americium by dibutylbutylphosphonate (DBBP) and diethylcarbamylnmethylenephosphonate (bidendate) extractants. The programs have been used to develop improved americium recovery flowsheets by evaluating the effects of the extractant, number of stages and the aqueous to organic (A/O) flow ratio. The results show americium losses were sensitive to the following in order of importance: (1) aqueous to organic flow ratio, (2) extractants and, (3) number of stages. The results are summarized below.

	<u>Extractant</u>	<u>Aqueous to Organic Flow Ratio</u>	<u>Stages</u>	<u>% Am Recovery</u>
1	DBBP	3.5	3	37
2	DBBP	3.5	5	38
3	DBBP	2.0	3	66
4	DBBP	2.0	5	72
5	Bidentate	3.5	3	51
6	Bidentate	3.5	5	55
7	Bidentate	1.6	3	84
8	Bidentate	1.6	5	93

Runs 1 and 2 represent the present level of americium recovery in the WE-1 Column at optimum pH. By replacing DBBP with bidentate and eliminating the acid neutralization step, americium recovery can be increased to 55 percent (runs 5 and 6). Further increases in americium recovery can be achieved by increasing the organic flow rate to lower the A/O ratio.

Americium Recovery Flowsheet - M. J. Klem

The purpose of this work is to develop a flowsheet for recovering and concentrating the americium from the Waste Treatment Facility solvent extraction process.

No further work has progressed on the preliminary design and scope for a new ion exchange system. Plans call for evaluation of ion exchange, thermal concentration, and precipitation processes for recovering and concentrating the americium.

A flowsheet document was issued for chromatographic ion exchange purification and conversion of americium to oxide.

Waste Treatment Columns - R. K. Owens, C. E. White

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

Previous pilot column studies using cerium(III) as a substitute for Am(III) verified the dependence of Ce(III) extraction by DBBP in CCl_4 organic on pH adjustment.

Preparation got under way at quarter's end to make pilot column tests of diethylcarbamylmethylenephosphonate (bidentate) as a potential replacement for DBBP. The bidentate is a better extractant for americium in high acid solution. Bidentate usage eliminates the high salt waste neutralization step necessary for the DBBP system.

2.3 SCRAP RECOVERY - Z PLANT

Canyon Floor Cleanup - J. A. Compton, C. E. White

The purpose of this work is to develop equipment for removing accumulated solids from the access glove box and canyon floors of the PRF.

Testing of a pilot vacuum cleanup system in the Engineering Development Laboratory has been completed. Documentation of the test activities is nearing completion.

A full-size vacuum cleanup system has been constructed and installed in the east side access glove box of the PRF. The system consists of a slurry transfer head, two glass tanks, a pump, filters, and related piping. All the equipment is geometrically favorable for handling the floor slurry. All the requirements of the Hazards Review Committee have been satisfied. Documentation of the process flowsheet is nearing completion.

Plutonium Scrap Dissolution - J. V. Gilchrist, C. E. White

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

No progress has been made on the critical mass evaluation of relocating the prototype dissolver to a glove box and using PuO_2 as feed. The project is a low priority item because of the Z Plant start-up requirements.

Reactive Scrap Disposal - J. T. Knight

The purpose of this work is to develop processes and equipment for stabilizing reactive plutonium scrap for storage or final disposal.

Resins. A process was developed and procedures prepared for converting cation exchange resin to a stable ash residue for disposal. The procedure involves processing small amounts of resin through a carbonization step for stabilization, followed by pyrolysis for volume reduction. This process was approved by the Hazards Review Committee and ten bottles of resin stored in 242-Z Building were processed. The facility has been left intact and will be used to process resin from the W-1 Column when it becomes available.

Rags. Disposal of contaminated combustible wastes at other ERDA facilities has been investigated and a trip report issued. A disposal process concept has been developed and work is in progress on equipment and facilities designs. An extensive literature survey has been completed on the effect of nitric acid on rags (cellulose) and a report is being prepared.

Sludges. Work is in progress to further characterize the sludges and to develop a method for separating plutonium for possible recovery.

Bidentate Extraction - W. W. Schulz

The purpose of this research activity is to develop a bidentate organophosphorus solvent extraction process for coextraction recovery of americium and plutonium from the extraction column raffinate (current acid waste solution) of the Plutonium Reclamation Facility.

Mixer-Settler Runs. Organic feed (SIF solution) containing 0.72M HNO_3 , 0.02 g/liter plutonium for final scheduled Am-partition (S1 column) mixer-settler runs was prepared by extended mixer-settler extraction column runs with actual current acid waste (CAW) solution. The extractant was 30 percent DHDECMP-perchloroethylene purified by ethylene glycol washing and Amberlyst A-26 (Rohm & Haas Company) resin treatment. Pertinent results of the Am-partition column runs were:

<u>Mixer-Settler Stages</u>	<u>Flow Ratio SIX/SIF</u>	<u>%Am Not Stripped</u>
4	1:2	13.4
4	1:3	26.2
3	1:2	13.4
3	1:3	18.4

(In all tests the SIX was 0.1M HNO_3). The results of the test were in good agreement with those expected from batch contract experiments, confirming that 75 to 90 percent of the ^{241}Am can be stripped as a small volume of aqueous product. Plutonium data for these runs are not available. The Am product (SIP solution) produced in two of the S1 column mixer-settler runs (4 stages) was analyzed for metallic

impurities. Emission spectroscopic analysis showed these product solutions to contain 20-50 mg Sn/liter, 20-30 mg Mg/liter, 20 mg Al/liter, 16 mg Na/liter, 16 mg Si/liter, and <50 mg Fe/liter; the concentration of all other metallic impurities was less than 10 mg/liter. These data show, as expected from previously obtained distribution data, that the DHDECMP process produces substantially "cleaner" Am product than does the presently used dibutylbutylphosphonate recovery scheme. Americium product from the latter process is grossly contaminated with aluminum and sodium.

Ion Exchange Concentration of Americium. The Am product from the bidentate process is typically 1.0 to 1.25M HNO_3 , while that from the present DBBP recovery process is only 0.25 to 0.35M HNO_3 . In present PRF operation, a bed of Dowex 50W cation exchange resin is used to sorb americium (and plutonium) from the DBBP process product stream to concentrate and partially purify it from sodium and other impurities.

Small-scale (3.0-ml bed) ion exchange runs were conducted to estimate the capacity of Dowex 50W resin for sorbing Am^{+3} directly from the DHDECMP acid Am-product solution. In one test, 320 column volumes (CV) of 1.0M HNO_3 -0.21 g Am/liter were downflowed at 25° C and ~4.5 CV/hour through a 3.0-ml bed of 20-50 mesh Dowex 50WX8 resin. The one percent breakthrough point was reached after passage of about 27 g Am/liter. In contrast, 446 CV of 0.25M HNO_3 -0.2 g Am/liter solution were downflowed through a similar 3.0-ml bed of Dowex 50X8 resin without any detectable breakthrough; the resin loading in this latter run was ~90 g Am/liter.

Results of these tests show that Dowex 50W resin will indeed sorb Am^{+3} directly from the high-acid product of the bidentate process. However, additional resin beds and/or

more frequent elution steps would be needed to provide resin capacity equivalent to what can be obtained with dilute (0.25M) acid feed. Provision of more resin beds may be more practicable than installation of equipment for acid destruction or for thermal concentration of the Am product.

Computer-Assisted Flowsheet Development. A modified version of the SEPHIS computer program has been used to successfully model the bidentate Am-Pu extraction flowsheet. The modified version employs as input only equilibrium distribution data from Am, Pu, and HNO_3 . After allowance for the inefficiency of actual mixer-settler stages, agreement between SEPHIS-calculated and experimentally (mixer-settler) determined Am and Pu recoveries for various flowsheet conditions was excellent. The modified program has been used to derive an optimum bidentate flowsheet for PRF use.

Future Work. The program milestone date of October 1, 1977, is on schedule for completing laboratory-scale development of the bidentate flowsheet. Prior to this time, we expect to complete final mixer-settler strip (S2) column runs, and to provide analytical and other assistance, as needed, in cold pilot plant test of the bidentate process. A final report summarizing results not already reported will also be started next quarter.

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

Current Acid Waste Flowsheet. The current acid waste (CAW) Strontium Recovery flowsheet will be combined with CAW cesium recovery. The complete CAW flowsheet will be similar to the "Tentative B Plant CAW Strontium and Cesium Recovery Flowsheet" recently issued. The final flowsheet will include changes resulting from pilot plant testing and Chemical Technology Laboratory (CTL) studies and should be completed in the third quarter, FY 1978.

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

The objective is to provide an accurate description of cesium ion exchange recovery through flowsheet documentation, testing ion exchange resins, and use of an ion exchange computer program.

Ion Exchanger for Removal of Cesium from CAW. Laboratory ion exchange column studies indicate slightly better removal of cesium is achieved with IAW in the pH 4.5 to 6.5 range than around pH 2. IAW flow rate to the column during loading had little effect on the breakthrough curve up to three column volumes per hour. Rare earths apparently do not absorb on Zeolon-900 (product of Norton Company) from IAW

solutions. Coabsorbed sodium was easily eluted from the column with either 0.15M $(\text{NH}_4)_2\text{CO}_3$ or 0.3M HNO_3 . Strong nitric acid did not satisfactorily elute cesium from the aluminosilicate. A strong $(\text{NH}_4)_2\text{CO}_3\text{-NH}_4\text{OH}$ solution is recommended for eluting cesium from the column.

Additional laboratory runs are planned to give a more accurate description of the proposed CAW flowsheet.

Concern has been raised on the ability of Zeolon-900 resin to withstand high radiation conditions during CAW processing. Currently, the Battelle-Northwest cobalt-60 source is being used to irradiate resin samples to determine the effects of high radiation. Final results will be available by August 1977.

Purex Sludge Supernate (PSS) Ion Exchange Flowsheet. The Purex Sludge Supernate (PSS) Ion Exchange Flowsheet has been recently published. This flowsheet, which covers the completed PSS cesium recovery process at B Plant, will be helpful in the final development of the CAW flowsheet and the development of the flowsheet for cesium recovery of Tank 101-AY supernate. Tank 101-AY is scheduled for processing through ion exchange at B Plant in the near future.

Tank 101-AY Supernate Ion Exchange Flowsheet Investigation. Preliminary laboratory work has been performed to aid in designing a flowsheet for removing cesium from Tank 101-AY supernate liquid. Batch distribution data indicate Zeolon-900 is the best ion exchanger for this material. Synthetic 101-AY solution will be run on laboratory ion exchange columns of Duolite ARC-359 (product of Diamond Shamrock) and Zeolon-900 next quarter.

Cesium Purification. During recent cesium purification processing, it has been difficult to purify the cesium product to meet encapsulation standards. It has been recommended that CTL investigate the cesium purification flowsheet to determine if improvements within the flowsheet can be made. Currently, CTL is organizing and defining the program. Actual laboratory studies will begin in the fourth quarter, FY 1977.

3.2 ENCAPSULATION

The purpose of this program is to provide process technology necessary to solve equipment and process problems existing in the Waste Encapsulation and Storage Facility (WESF).

Tilt-Pour Cesium Equipment - J. D. Moore

Development of the tilt-pour melt caster continued with four weeks of successful production and reliability testing with the equipment. Operations were conducted remotely with hot-cell manipulators in a cell mockup to simulate WESF conditions.

The tests, 16 batches of nonradioactive cesium chloride were processed filling 112 capsules. This production rate easily exceeds that required to meet the WESF production goal of 500 capsules per year. No equipment or mechanical problems were experienced with the tilt-pour equipment. Melter crucible corrosion resistance is considered satisfactory for use in the WESF.

Currently, several minor design modifications to enhance equipment maintenance remotability are being completed. Remote assembly tests with the equipment will be carried out by September 1, 1977 before turning over the tilt-pour melt caster to WESF operations.

Alternate Compounds for Encapsulation - D. M. Strachan

Compounds which are more compatible with long-term waste management plans are being sought as alternates for encapsulation. During the past quarter, strontium compounds in the $\text{SrO-Al}_2\text{O}_3\text{-SiO}_2$ phase system were investigated. Acceptable leach rates in the low $1 \times 10^{-5} \text{ kg Sr}/(\text{m}^2 \cdot \text{s})$ [$1 \times 10^{-4} \text{ g Sr}/(\text{cm}^2 \cdot \text{d})$] were obtained with an $\text{Al}_2\text{O}_3/\text{SiO}_2$ mole ratio of 1.0 and up to 70 mass percent SrO .

Several samples of the CsCl -clay reaction product were sent out for microstructure analysis. Recent chemical analysis shows that there is 40.4 mass percent Cs_2O in these samples. For the particular clay which was used to make these samples, the stoichiometric amount of Cs_2O required was 42.0 percent. This suggests that less than 4 percent of the cesium was lost by volatilization during processing.

Hot-Pressing Strontium Fluoride - D. W. Jeppson

An alternative process for encapsulating strontium fluoride is being developed. Test pressings have been made to determine the optimum temperature and pressure combination to produce a high-density strontium fluoride product which can be easily removed from the mold. A molybdenum mold was used during the test pressing to allow the attainment of higher pressures. Pressing at a low temperature prevented the low melting eutectic formation and oozing problems experienced last quarter. The use of a graphite foil liner allowed the reuse of the molybdenum mold. One test pressing made at 650°C and 10,000 psi looked promising, but gave only 75 percent of theoretical density. Other pressings are being made at constant temperature while increasing the pressure until an acceptable density is obtained.

3.3 WASTE CONCENTRATION

The objectives of this program are to transform 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.

Using the Heat-4 computer code, steady-state temperatures were calculated for Tank 104-AX, assuming that there was a six inch deep layer of sludge on the bottom of the tank. Other assumptions for the calculation were: (a) the volumetric heat generation for the sludge is 60 Btu/hr-ft³, (b) the sludge is not covered with salt cake (which may contain volatile hydrocarbons), and (c) there is no ventilation of the air space in the tank. The maximum tank bottom temperature was calculated to be 727° C. This mean air cooling must be provided or else five inches of the sludge must be removed.

Interstitial Liquor Removal - W. C. Schmidt, R. E. Van der Cook

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.

Removal with Jet Pumps. One of the prototype pumps (TK-107-BY) operated 68 percent of the time and removed 7,900 gallons of liquor. The pumping rate was slow and varied from 0.16 gpm after hot water backflush of the pump well screen to 0.08 gpm (115 gpd). The major operating problem was plugging of the transfer line due to heat tracing failure. The other prototype (TK-111-S) did not operate during the quarter due to needed maintenance work and work priority. Repairs were completed at quarter's end; however, interferences with pumping transfers to obtain evaporator feed and construction of the S-SX jet pump project will limit TK-111-S operations during the next quarter.

Design and equipment modifications were recommended for the upcoming jet pump installations. The 1/2-inch magnetic flow-meter for the S-SX project was tested and found to be accurate below manufacture literature down to a rate of 0.05 gpm. An S-SX jet pump unit has been assembled for pre-operational performance testing.

Reduced Pressure Pump. Discussions were held with a consultant, Dr. Lyman Handy, on capillary and reduced pressure theory. Since then, radial flow equations were used to model the reduced pressure system of vacuum-porous metal-diatomaceous earth-salt cake. One model indicates that the system will not work but will suck the liquid down near the pump well. The other model assumes the theory would work and predicts flow rates of 10-30 gpd with diatomaceous earth and 100-200 gpd without. Experimental porous metal tubes have been received and laboratory experiments are being started to determine if the system is feasible and what rates are indicated.

In-Situ Drying. Laboratory studies on drying were initiated in early June 1977. These studies will be used to evaluate empirical constants necessary to estimate drying times. In addition to laboratory studies, a process test is planned for July 1977. The process test on selected waste tanks will measure drying rates versus air flow rate. The results of this test will aid in finding the maximum drying rate of the tank solids. Future work in in-situ drying will be determined from the results of these studies. Improved methods for taking psychrometric data are being explored.

Ostwald Ripening. Analysis of the laboratory data from the previous experiments to establish rate constants for inclusion in a mathematical model did not produce any significant results. A revised experiment utilizing a bimodal particle size distribution is being planned to obtain the necessary data.

Residual Liquor Disposition - R. E. Van der Cook,
J. O. Honeyman

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.

Double-Shell Slurry Process. The double-shell slurry process test was run from April 25 to April 28, 1977. Approximately 364,000 gallons of Hanford Defense Residual Liquors with an aluminate-to-caustic molar ratio of 0.6 were reduced to 271,000 gallons of double-shell slurry for an average volume reduction of 25 percent. No unusual problems were experienced during routine operation. The slurry was pumpable, although

the slurry specific gravity was slightly above the targeted value (1.72 versus 1.68). Product evaluation is continuing.

Additional pilot plant runs were conducted in late March 1977. The pilot evaporator-crystallizer was operated for several days at a high aluminum supersaturation to determine if any evaporator fouling occurred. Fouling occurred at the vapor-liquid interface, as expected, with a small amount of general body fouling observed after a day's time. The fouling was easily removed and at no time did solids deposit on the heat transfer surfaces.

Partial Neutralization. The partial neutralization prototype equipment has been accepted from construction by the Operations Department with a few exceptions. Operational Test Procedures are currently being conducted with an anticipated startup date of early August 1977. Two documents on this subject were issued during the quarter.

Salt Cake and Sludge Characterization - G. K. Allen

The objective of this subtask is to obtain core samples of waste tank solids, to provide chemical and physical properties of the solids for the support of long-range radionuclide removal and solidification studies, to evaluate the effectiveness of the salt well program, and to provide data to numerous smaller products.

Additional cold testing of the drilling equipment was conducted to determine the causes of low core recovery in the 241-TX-116 hot test. Test results showed that several new pieces of equipment and different operating methods are required to recover a high percentage of representative cores

from Hanford waste tanks. Required equipment includes a new drill rig, diamond drill bit, chrome plated sample tube, additional transfer casks, and an enclosure allowing continuous coring during windy periods. New operating methods include the use of a professional drilling operator to operate the drill rig and the continuous coring of the solids within the waste tank.

The new drill rig will provide a vibration-free drill string rotation, measurement of the downward drilling force, and the speed of the rotating drill bit. The diamond drill bit and chrome plated sample tube will increase core recovery significantly over the previous carbon steel drill bit and non-coated carbon steel sample tube. Steel penetration tests conducted on the diamond drill bit showed that tank liner penetration could be controlled to a negligible level when applying drilling forces required to core through moist sodium nitrate salt cake. Continuous coring of a tank is required to prevent hole cave-ins around the drill rod. Cave-ins cause a loss of drilling air circulation which plugs the drill bit. Continuous coring of a tank will require: (a) additional transfer casks for lag storage at the drill site, and (b) a wind shield which allows drilling during windy periods. The new equipment and drilling methods should enable the effective drilling and recovery of continuous cores from Hanford waste tanks.

3.4 TANK FARM SURVEILLANCE AND MAINTENANCE

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

The Boeing Company has developed a prototype soil conductivity waste tank leak detection system under a follow-on contract resulting from a previous concept research contract. The prototype installation is now complete with five selected tanks located in 241-A and 241-AX Tank Farms being instrumented. A portable system installed in an automotive van has also been developed. Final documentation delivery and contract completion was accomplished June 8, 1977. Final billings have not been received, but the estimated costs ending this development phase total \$460,000; distributed \$357,000 for Boeing, \$99,800 for J.A. Jones, and \$3,200 for Direct and Applied Materials. The systems are currently operational and are being evaluated. The final systems disposition recommendation is scheduled for August 1, 1977.

With configuration modifications, Boeing is adapting the leak detection system for transfer pipeline applications. Estimated cost for this effort is \$7,000 for Boeing and \$5,000 for an ARHCO test facility. The design is now complete and testing is expected to start by August 15, 1977.

A demonstion "In-Place" gamma monitoring system has been developed to provide computerized surveillance of waste tank leak detection dry wells. This system provides an alternative to the soil conductivity method, if one should be required.

An ultrasonic flowmeter has been tested for possible use in conjunction with radioactive waste transfers and salt well pump flow measurements. Although the device has not proven to be desirable for these uses, there are many other applications where the portability and quick installation capability of the unit can be utilized by plant forces. The evaluation unit is now being assigned to operations for use as required.

A soil moisture monitoring detector (neutron probe) for scanning waste tank leak detection dry wells is being developed. Preliminary testing of the neutron source geometry and detector (BF-3 tube) has been completed. These tests show greatly improved resolution as compared to the existing commercial probe and no gamma response in the wells tested (approximately 30 r/hr). This work is now about 90 percent complete.

4.0 LONG-TERM MANAGEMENT OF DEFENSE WASTE AT HANFORD

(D. R. Gustavson - Program Manager)

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 2,900 capsules of ^{90}Sr and ^{137}Cs stored in water basins and 50,000,000 gallons 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 the biosphere. 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:

- o To retrieve the waste, process it to an improved stabilized form, and store it in deep natural rock or other formations (geologic storage).
- o 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.
- o 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.

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

The objectives of this program are:

- o To develop data to facilitate end-of-life projection for tanks.
- o To develop a data base to facilitate the continued assessment of the safety of continued storage of solidified waste in the existing single-shell tanks; to be completed in FY 1980.
- o 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.
- o 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.

Storage System Integrity

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

Salt Cake Stability - G. A. Beitel

The nitrates which form the bulk of salt cake may be theoretically capable of reacting with any organic materials which may be present. Tests have shown no evidence of susceptibility to shock-initiated detonation or to combustion within the moisture-nitrate-organic mixture calculated to be present.

During this past quarter, a series of pressurized deflagration tests was conducted by Hazards Research Corporation, Denville, New Jersey. Samples of simulated salt cake, kerosine, B Plant complexant, and resin were contained in a sealed capsule and heated to 250° C. Both pressure and temperature were monitored during the heat cycle. Several samples were heated to 450° C in an unpressurized container and several unheated kerosine mixtures were ignited. The complete details of these tests are described in a recent report which is now in the process of being cleared for release. This document satisfies a program milestone and is on schedule.

These, and prior tests under this program lead to the following conclusions:

1. Sodium nitrate based salt cakes cannot be made to detonate regardless of the type of organic material added, provided, of course, that the organic added is not detonable by itself.
2. Sodium nitrate based salt cakes containing organic materials are stable against deflagration initiation for temperatures below 250° C.
3. Sodium nitrate based salt cakes will ignite and react explosively with most organic materials which are still liquid or solid at temperatures above 400° C. Ignition will occur between 400 and 450° C.
4. The energy content of a sodium nitrate based salt cake with less than 5 weight percent oxidizable organic is too low to support combustion even if ignited.

5. If a volatile hydrocarbon is added to a sodium nitrate based salt cake and subsequently heated, the hydrocarbon will evaporate unreacted before ignition temperatures ($>400^{\circ}\text{C}$) are reached.
6. If a volatile hydrocarbon is added to a sodium nitrate based salt cake and subsequently ignited, the hydrocarbon will flame above the salt cake, reacting with environmental gaseous oxygen (assuming an adequate source is available). The salt cake will act as a wick, transporting the hydrocarbon to the surface by capillary action and evaporation will cool the salt cake to below ignition temperatures.

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 elevated temperature, thermal cycling, and chemical solutions have affected the elastic properties of the concrete and the reinforcing steel.

o Elastic Properties

Tests are continuing at the Portland Cement Association (PCA) to determine long-term elevated temperature effects on the elastic properties (compressive strength, modulus of elasticity, tensile splitting strength, and Poisson's ratio) of the concrete used at Hanford. These tests are scheduled for four years. The data for nearly two years of testing are available. Attached are plots of the modulus of elasticity and compressive strength data obtained after exposure at 250°F (see Figures 4-1, 4-2, 4-3 and 4-4). Although the compressive strength showed

FIGURE 4-1. Compressive Strength

FIGURE 4-2. Modulus Elasticity

FIGURE 4-3. Compressive Strength

FIGURE 4-4. Modulus Elasticity

an initial decrease, it remains significantly higher than the minimum specified strengths of 3000 psi and 4500 psi.

o Tank Dome Cores

Concrete cores removed from tank domes because of operational modifications have been shipped to PCA and cut into specimens for elastic property testing. The program requires testing at 70° F and 250° F. The 70° F test data are tabulated below. The concrete had a specified minimum compressive strength of 3000 psi and a modulus of elasticity of 3.5×10^6 psi.

<u>Compressive Strength (psi)</u>	<u>Modulus of Elasticity (10^6 psi)</u>	<u>Poisson's Ratio</u>	<u>Tensile Split Strength* (psi)</u>
5190	3.776	0.187	850
5025	4.070	0.193	769
7702	4.833	0.222	1018
8237	5.322	0.191	1255

*Separate specimens required for tensile split tests.

o Coefficient of Thermal Expansion

Some coefficients of thermal expansion data have been obtained for 3000 psi minimum compressive strength Hanford concrete. Four thermal expansion tests have been completed for a temperature range from 25° C (77° F) minimum to 500° C (932° F) maximum. The following is a table of the coefficient of thermal expansion values calculated from the data. These are average values in the temperature ranges shown.

<u>Coefficient of Expansion</u> <u>(in/in - °F)</u>	<u>Temperature Range</u> <u>(°F)</u>
4.44×10^{-6}	77 to 212
3.80×10^{-6}	77 to 392
3.89×10^{-6}	212 to 572
3.17×10^{-6}	212 to 536
3.90×10^{-6}	77 to 691
3.40×10^{-6}	392 to 691
2.60×10^{-6}	536 to 752
3.33×10^{-6}	572 to 752
3.06×10^{-6}	752 to 932
2.22×10^{-6}	752 to 932

The value for the coefficient of expansion of concrete generally given by the literature is 6.0×10^{-6} in/in °F.

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.

Bench marks were attached at selected locations to tanks in the A and AX Tank Farms. The first set of elevation measurements has been obtained. Measurements will continue to be made on a quarterly basis to determine the magnitude of dome elevation change.

Tank Safety Analysis - R. G. Baca, P. G. Easley

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

Moisture and Radionuclide Modeling

A consultant contract was finalized with Intera Environmental Consultants, Houston, Texas. Work was initiated by Intera on the analysis of three tank failure modes: (1) liner crack, (2) sidewall failure, and (3) dome collapse. In this computer simulation study, a variety of leak scenarios will be analyzed using state-of-the-art computer models. Simulations of moisture and radionuclide movement through the Hanford vadose zone will be performed. These results will provide the information needed to assess the potential for contaminant transport to the groundwater and to the ground surface. The study will focus on the near-term dynamics of these transport processes. Work on this task is on schedule.

Hydrogeochemical Studies

Bids were solicited from qualified consulting firms for the conduct of the hydrogeochemical studies. A contract was awarded in July 1977 to the Institute for Research, Houston, Texas. These studies will be completed early next fiscal year; the delays encountered in initiating this task will not impact on the major milestone for FY 1978. This task is approximately three months behind schedule.

Airborne Transport Modeling

A consultant contract with Kaman Sciences Corporation, Colorado Springs, Colorado, was processed. The staff of Kaman is working on a computer simulation study of postulated explosions in a subsurface single-shell tank. A sophisticated computer code is being used to simulate the fracture, deformation, and displacement of tank and earth materials for various explosion scenarios. The theoretical analysis will consider different explosion mechanisms and magnitudes.

The results of this study will provide a technical data base needed for assessing the potential of venting salt cake materials for postulated accidents and sabotage actions. A 35-mm movie will be made which will show the time-history for each explosion scenario.

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. Testing and selection of engineered improvements to the storage tanks are scheduled for completion in FY 1980.

Conceptual studies of engineered improvements for the waste storage tanks have led to the conclusion that filling the tank dome void space may offer a method to provide support for the dome at such time that it may be needed. A program plan will be developed which will include evaluation of the effect of filling the space on radiolytic heat removal and hydrogen generation. Also to be included is determination of a filler material that is chemically inert with the waste and does not preclude retrieval in the future.

4.2 WASTE RETRIEVAL - H. A. Wallskog

The objective of this task is to develop and demonstrate a retrieval system which is capable of retrieving material from Hanford waste tanks in a safe, efficient, and cost effective manner. To achieve this objective, a plan has been formulated.

which calls for the design, fabrication, testing and demonstration of a functionally complete, full-scale Prototype Retrieval System capable of retrieving solidified waste from any of the large, underground waste storage tanks. The principal goal of this technology development task is a radioactive demonstration of the system in one or more of the Hanford waste tanks, presently scheduled for FY 1983.

Conceptual Design - Prototype Retrieval System -

H. A. Wallskog

Work on the system conceptual design by Programmed and Remote Systems Corporation (PaR), St. Paul, Minnesota has been completed. The final report was received May 31, 1977. The PaR report will be reviewed and used as a basis for the "Conceptual Design - Prototype System Study Report" to be issued in September 1977. A contract has been issued through PaR to Boucher-Lewis Precision Models, Inc., Minneapolis, Minnesota, for the fabrication of a manually operable scale model of the Prototype Waste Retrieval System. Established on June 15, 1977, this contract is scheduled for completion early in the first quarter of FY 1978. Designed to be transported in small subsystem components, the model will be used for demonstration purposes and to analyze the Prototype Waste Retrieval System.

Tool and Equipment Development - N. M. Arnold

"Slurry Elevator". A purchase order has been issued to Wilkins and Associates, Inc., Tacoma, Washington for the fabrication and testing of the Prototype Slurry Elevator. Detailed design analysis will begin in early July 1977 with project completion date expected by the end of the first quarter of FY 1978.

"Scumbuster Test" - Model 150. Damage to a piece of test equipment (flow meter) has caused another delay in the completion of this test. The replacement component is expected to arrive from the factory by mid-July. However, rescheduling of this test is not firm due to the commitments of time and personnel to higher priority efforts.

Engineering Studies and Analyses - N. M. Arnold, D. N. Morrill, J. K. Strickler, and H. A. Wallskog

Analysis of the tank dome structure for the addition of two 42-inch diameter penetrations, one at the tank dome center and one at a radius of 20 feet 9 inches, has been completed by Dr. V. B. Watwood, Houghton, Michigan (Contract CA-603). The final report (ARH-C-19, May 24, 1977) concluded that the stresses in the tension ring of an uncracked concrete dome will increase by 10 to 15 percent. The report also stated that, if the concrete has sufficient strength and ductility, failure due to addition of these penetrations will not occur. Dr. Watwood recommends that additional concrete cores be taken from the domes of the 241-U tanks, which are generally considered to be the weakest due to their age and concrete specifications at the time of construction, and tested to ensure that the concrete has not embrittled over the years.

The engineering study, defining the site preparations required before a Waste Retrieval System could begin waste retrieval operations in 241-SX Tank Farm, has been completed and will be documented. Documentation of the shielding study made for three different shipping containers has been completed.

Although most of the waste is stored in the large seventy-five foot diameter tanks, there are more than one-half million gallons stored in smaller twenty foot diameter tanks. Because

the Prototype Retrieval System is not designed for the smaller tanks, a study was initiated to determine a satisfactory means to retrieve this waste.

4.3 WASTE PROCESSING - R. A. Watrous

The objectives of this program are:

- o To identify, develop and demonstrate the technology for converting all defense high-level waste to an immobile form.
- o To select a reference process for waste processing and demonstrate its technical and economic feasibility.
- o To analyze and evaluate alternative processes for waste processing in relation to the reference process.
- o To design and develop equipment, pilot plant and demonstration facilities for proving the selected process.

Immobilization Processes - Laboratory Development

The objectives of this activity are to develop and demonstrate processes for incorporation of Hanford high-level wastes into immobile silicate-based glass forms, alumino-silicate ceramic forms, or low-temperature cement or grout-based forms.

Silicate Melt Process - Formulation Development - M. J. Kupfer

During the period July-September 1977, pilot scale melting tests will be performed using the Battelle-Northwest joule-heated and in-can melters for conversion of simulated Hanford defense wastes to glass.

In preparation for these tests, several waste-borosilicate glass formulations were developed, three of which were chosen

for the melter demonstration tests. The simulated wastes to be immobilized in these tests are representative of those produced both with and without "Radionuclide Removal" processing. Very low melting glasses were chosen for the tests because of the maximum temperature limitation of the existing BNW melters ($\sim 1100^{\circ}\text{C}$). Thus, durabilities of the resulting glasses will be relatively poor compared to the higher melting ones, and are probably not representative of those that will be chosen for final fixation of Hanford waste.

Documentation of Silicate Melt Studies - M. J. Kupfer,
W. W. Schulz

A topical progress report on the Silicate Melt immobilization of Hanford defense wastes has been documented and issued as ARH-SA-285.⁽¹⁾ The report summarizes glass formulation and characterization tests performed in the ARHCO Chemical Technology Laboratory from early 1975 through March 1977.

Clay Calcination Process Development - D. M. Strachan

During this quarter, work continued on the clay calcination process using synthetic wastes typical of average Hanford waste [salt cake plus sludge (SC)] and low-volume wastes resulting from the radionuclide removal process. These studies yielded preliminary data which suggest that leach rates less than $10^{-5}\text{g/cm}^2\cdot\text{d}$ (based on Cs) are possible for products containing as much as 62 percent (with radionuclide removal) waste oxides. Leach rates based on Sr were even lower. Similar data were collected for the SC products. For the SC products, the cesium leach rates remained low even though the sodium leached rapidly. Preliminary data also indicate that wet milling the waste with the clay prior to calcination yields lower leach rates for both cesium and strontium.

Densification of the clay calcination products was studied, but not exhaustively. By wet milling the calcined waste-clay mixture with acetone or in a mixture of 79 percent clay-waste, 10 percent acetone, 10 percent water, and one percent stearic acid, pellets could be pressed at 13.6 Mg (30×10^3 lb) instead of 2.27 Mg (5,000 lb) using a 22-mm (nominal one inch) die. Pellet green densities (densities prior to firing) increased 30 to 40 percent and fired densities increased about six percent.

All of these data will be verified during the next quarter.

Immobilization Processes - Equipment Development

The objectives of this program are to develop and demonstrate processing equipment and technologies for converting Hanford high-level waste to an immobilized form satisfactory for long-term or final storage. Current emphasis is primarily on development of glass melter equipment. Pending the outcome of alternative product form evaluations, equipment development for the clay calcination or grout immobilization processes may receive increased emphasis.

Melter Development Test Plans - L. A. Chick

Work scheduled at BNW includes: (1) test runs of simulated defense waste glasses using equipment developed for the commercial waste immobilization program, and (2) construction and operation of a bench-scale melter specifically for the defense waste program.

In-Can Melter Tests are scheduled for late June and July 1977, with the objective of providing the first evaluation of the technical applicability of the in-can melter process to immobilization of Hanford defense waste after radionuclide

removal. Two typical radionuclide removal glass compositions will be run in the pilot scale calciner and in-can melter to produce two eight-inch diameter by sixty-inch high Inconel 601 canisters of glass product. Completion of this work on schedule will depend upon timely delivery of chemicals.

Continuous Ceramic Melter (Tilt-Pour) Tests are scheduled for July and September 1977, and should provide the first evaluation of technical applicability of the continuous ceramic melter process to immobilization of Hanford defense waste. Two glass formulations developed by CTL will be run in the continuous ceramic melter, simulating low or high sodium-nitrate waste feeds (i.e., with or without the radionuclide removal feed pretreatment option). These runs will produce four 24-inch diameter by 8-foot high mild steel canisters of product. Completion of these tests also depends upon delivery of chemicals.

Battelle-Northwest has begun design work and ordering of crucial materials for a bench scale melter. The two major objectives for this melter will be: (1) to evaluate refractory and electrode materials and configurations, and (2) to evaluate the effects of waste feed variability on melter operability and component life. Plans are to build a small (20-40 pounds glass per hour) melter to be operated for about two years, testing many different possible defense waste glasses. During the program, the melter will be rebuilt several times to evaluate candidate refractory materials and electrode materials including Inconel, tin oxide, and molybdenum.

Materials Corrosion Tests at Penberthy Electromelt Company -
L. A. Chick

Scouting materials corrosion tests were performed at Penberthy Electromelt Company in Seattle during March 1977. Five candidate refractory and electrode materials for the bench scale melter were heated in four different simulated defense waste glasses. The test matrix and glass compositions are shown in Tables IV-1 and IV-2.

The purpose of these tests was to determine if any of the candidate materials would experience catastrophic failure when exposed to an environment of molten waste glass compositions.

TABLE IV-1

Test Matrix For Melter Materials Corrosion

Glass Number	1	2	3	4
Glass Type	Average RR No uranium	Average RR W/uranium	Average W/O RR High nitrate	Average W/O RR Sod. Carbonate
Test Material	Sample #	Sample #	Sample #	Sample #
Tin Oxide	1	2	3	
27% Chrome Refractory Mat'l	4*	5*	6*	
	7	8	9	
Platinum	10	11		
Molybdenum	15	13	14	12
Incone1 690	16	17	18	

*Note: Samples 4, 5, and 6 held at 1400° C for 48 hours.
Samples 7, 8, and 9 held at 1200° C for 48 hours.

TABLE IV-2

Composition of Glasses Tested for
Melter Materials Corrosion

Oxide	Substitute	Glass #1	Glass #2	Glass #3	Glass #4
SiO ₂		147.60	143.10	167.40	167.40
Na ₂ O	Na ₂ CO ₃	71.98	69.73		133.30
	NaNO ₃			213.80	
B ₂ O ₃			43.80 ⁽¹⁾		
	B ₂ O ₃ ·3H ₂ O	79.92	79.92 ⁽²⁾	54.35	54.35
Li ₂ O	Li ₂ CO ₃	14.81	14.37	14.81	14.81
Al ₂ O ₃	Al ₂ O ₃ ·3H ₂ O	32.11	31.92	12.38	12.38
Fe ₂ O ₃		10.50	10.20	1.20	1.20
SrO	SrCO ₃	10.17	9.70	0.05	0.05
P ₂ O ₅		5.01 ⁽³⁾	4.86 ⁽³⁾	6.63 ⁽³⁾	6.63 ⁽³⁾
Bi ₂ O ₅		3.00	2.91	0.30	0.30
CeO ₃		2.94	2.85	0.30	0.30
ZrO ₂		2.79	2.70	0.33	0.33
Ni ₂ O ₃		2.07	2.01	0.24	0.24
MnO ₂		1.74	1.68	0.18	0.18
CaO		1.68	1.62	0.18	0.18
Cr ₂ O ₃		1.29	1.26	0.45	0.45
TiO ₂		0.60	0.57		
WO ₃	H ₂ WO ₄	0.15	0.15	0.03	0.03
CdO		0.06	0.06		
UO ₃			9.00		
Totals:		388.42g	386.48g	472.63g	392.13g

- Notes: (1) 73.80g B₂O₃ used in glass type 2 with refractory samples No. 2 and No. 5.
- (2) 77.79g B₂O₃·3H₂O used in glass type 2 with all other refractory samples.
- (3) Error of approximately ±0.15g due to difficulty in handling material.

All samples were placed in kyanite crucibles with the appropriate waste glass and heated to 1200⁰ C in an electric radio-element furnace. They were held at temperature for 48 hours, removed, drained or quenched, cleaned, and measured and observed for loss of material via corrosion.

With the exception of Sample 12, molybdenum, no samples exhibited significant measurable evidence of excessive wear or catastrophic failure. The failure of Sample 12 was due to accidental exposure of the molybdenum to air at temperature, resulting in oxidation of the portion protruding above the glass.

From the test results, it is reasonable to conclude that the materials and glass compositions tested would be compatible for use in a continuous melter. It should be noted, however, that accelerated wear will take place in the presence of an electric field, particularly at energy-transfer interfaces. Longer term testing of these materials in the bench scale melter will give the best information in regard to life expectancy in service conditions.

Glass Canister Development - Hot-Pour Tests - R. C. Lawrence
(BNW), G. W. Grimes

Two canisters typical of those being considered for encapsulation of defense waste glass were tested at Northwestern Glass Company, in Seattle, Washington, on April 28 and 29, 1977, under the direction of BNW personnel. The test objective was to evaluate the dimensional stability and, hence, acceptability of a relatively simple canister design under conditions of rapid hot-pour filling and natural cooling (i.e., without controlled annealing). Prior to the testing of these canisters, it was believed that an insulating barrier would be

necessary to protect the steel canisters from excessively high temperatures which would cause canister failure.

Five-foot tall canisters were fabricated from two-foot diameter, schedule 80 (3/8-inch thick), carbon steel pipe with ASME-approved tank heads welded to the ends of the pipe. The canisters were filled (via a 6-inch diameter filler neck) with "coke bottle" glass having a density of 150 to 170 pounds per cubic foot. The glass entered the canisters at a temperature of 2150° F and at flow rates averaging approximately 80 and 150 tons per day. The highest temperature measured on the outside canister walls was 1200° F. Inspection of the canisters after the glass pour revealed no visible bulging or areas of stress failure; however, destructive testing remains to be done. Further evaluation of the uninsulated canisters may prove the insulating barrier to be unnecessary.

Glass Formulator Code - G. W. Grimes

Development was begun on a computer program which can assist in calculating glass formulations by modeling the conversion of wastes (as they are processed through radionuclide removal and melting) to waste oxides. Ultimately, the program could lead to the prediction of glass properties. The flow diagram for the program is shown in Figure 4-5. Those portions of the program which are complete and functioning are shaded.

At present, the program is an interactive tool for rapidly designing glasses from actual wastes. When sufficient laboratory data on waste glass characteristics and tank waste compositions become available, the program will be expanded to design waste glasses, given only the desired glass characteristics and a waste tank number or waste composition.

FIGURE 4-5. Glass Formulator Program Flow Diagram

Radionuclide Removal Process

The objective of this activity is the identification, development, and demonstration of technology for removal of all long-lived (half-life >25 years) radionuclides from Hanford salt cake and residual liquor. The process would partition 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 relatively inexpensive means. Reduction of total processing and storage costs is the incentive.

Sludge-Salt Cake Separation - M. J. Kupfer

One early processing step in the overall radionuclide removal process requires development of the technology to physically separate the water insoluble sludge from the salt cake solution. (The sludge would then be combined with the concentrated radionuclide fractions, resulting from ion exchange processing of the salt cake solution, and converted to immobile glass or crystalline forms suitable for long-term storage.)

During the report period, laboratory scale tests were conducted to determine the experimental conditions necessary to provide effective separation of actual salt and sludge mixtures obtained from four different Hanford tanks. Two washings (two parts by weight water to one part waste) reduced the soluble salt content of sludge-salt mixture to the range of 4-20 weight percent, depending on the initial level of salt contamination. Based on visual observations, peptization of the sludges did not occur during any of the washing steps. An important question still remains unanswered, however. How much (if any) sulfate is associated with the waste, and how many washing steps are necessary to remove sulfate from the

sludges? Although there is presently no reliable procedure in place for analysis of sulfate in sludge, development of a new method is nearing completion.

Kinetics of Ozonolysis Reaction on Complexants - T. W. Lutton

Previous work has shown that complexants, which inhibit the sodium titanate sorption of ^{90}Sr and actinides, can be oxidized to innocuous species by treatment with ozone at 60-90° C. Current work is aimed at determining detailed knowledge of the kinetics and other characteristics of ozonolysis of ethylenediaminetetraacetate (EDTA) and other complexants.

Work during the report period involved:

- o a review of analytical methods for determining low concentration of EDTA,
- o acquiring and setting up equipment to perform required analysis, and
- o setup and testing of laboratory ozonation equipment.

Results of scouting runs to test equipment and proposed analytical methods are plotted in Figure 4-6. Although very preliminary, these results show that ozone at 60° C oxidizes uncombined EDTA at easily measurable rates. The linearity of the plots in Figure 4-6 indicates not only that the rate of reaction is fast, but that the reaction rate very likely depends upon the rate at which ozone can be supplied to the EDTA. The steeper slope of the curve for Run 2 shows the ozonolysis rate is pH dependent, possibly reflecting increased aqueous phase solubility of ozone at higher pH's.

Oxyphotolysis of Complexants - P. C. Ely, T. W. Lutton

A promising alternative to simple ozonation of organic complexants is to oxidize them with a combination of ultraviolet

FIGURE 4-6. Ozonation of a 0.1M Na_2EDTA Solution

light and ozone. Such a combination is the basis for the Oxyphotolysis process patented by Houston Research, Inc., and used by them to destroy cyanides in waste waters. We are currently negotiating with Houston Research to execute a statistically-designed experiment to test effectiveness of combined ozonolysis and ultraviolet light irradiation in destroying organic chelating agents in Hanford alkaline waste liquors. We have also ordered a custom-made photo-chemical reactor (Ace Glass Co.) as well as various lamps and power supplies.

Development of Sodium Titanate Ion Exchange Media - C. A. Knox

Sandia Laboratories, Albuquerque, New Mexico, (R. Dosch) continued development and testing of sodium titanate to be employed as an ion exchanger for removal of strontium in the radionuclide removal process. Current development objectives are to consolidate sodium titanate powder, determine the effects of consolidation on titanate chemistry, and column test the powder and pellets with a dissolved salt cake simulant.

Previously reported binders were Cerma-Bind 500, water and epoxy. The Cerma-Bind pellets, which had favorable strontium sorption rates, were found to disintegrate when the column was loaded. Attempts to improve sorption rates with the epoxy binder failed. Therefore, the pellets with the water binder were selected for the column work. Several new binders have been tested, but only two showed promise - a water miscible glue and polyethylene powder. Column runs were made with water binder pellets to get an indication of the performance of pelletized material in removing trace levels of Sr. Two columns were operated in series (one with pellets, the second with crushed pellets) until 3000 column volumes of feed

had been put through the columns. The packing efficiency of the first column was poor.

The C/Co* for ^{85}Sr for this column increased slowly to a final value of 0.32. The overall decontamination factor for the two columns in series was a nearly constant 5000 throughout the experiment.

Sandia Laboratories modified their sodium titanate powder loading procedure to rectify previous plugging problems. The column was loaded and has been in operation for two months. There has been a one percent bed shrinkage, no appreciable change in flow resistance after startup, and the only sign of bed deterioration has been a small fingering pattern at the top edge of the bed.

Pilot Plant Hydraulic Testing of Powdered Titanate Exchanger -
B. A. Bell, C. A. Knox

The Waste Management Pilot Plant (202-S Building) is also investigating the hydraulic flow characteristics of the titanate ion-exchange resin developed by Sandia Laboratories. Preparation for pilot scale, hydraulic flow testing of a powdered titanate column (four-inches diameter by five-feet high) was completed and testing begun. A diagram of the test stand is shown in Figure 4-7. Synthetic redissolved salt cake solution is pumped through the four-inch diameter titanate column at various flow rates and the hydraulic pressure drop across the bed is recorded. Additional related variables to be investigated include: bed height, column backflow, and resin fines loss. Results of this pilot plant activity should determine the feasibility of using titanate resin in its presently produced (powdered) form.

*C/Co denotes the column breakthrough factor (curies ^{85}Sr in effluent/curies ^{85}Sr in feed).

FIGURE 4-7. Schematic Diagram - Hydraulic Test Column

Sorption of Strontium-90 on Sodium Titanate - W. W. Schulz

Current Radionuclide Removal flowsheets envision use of a combination of inert strontium phosphate scavenging and/or sorption on sodium titanate exchanger to remove ^{90}Sr from feed liquors. The large capacity of the titanate exchanger for sorbing strontium has been demonstrated by Mr. R. Dosch at Sandia Laboratories, with simulated feedstock spiked with ^{85}Sr . A capacity test is now in progress in our laboratories using a composite of various actual salt cake solutions, near residual liquors, and unevaporated tank solutions.

Phase 1 of this "hot" capacity test has been completed. This phase involved passage of 1510 column volumes (CV) of feed solution downflow through a 2.35-ml (0.7-cm diameter) bed of titanate exchanger at 25°C and $\sim 6\text{ CV/hr}$. Important feed constituents included 4.42M Na^+ , 0.91M OH^- , $3.2 \times 10^4\text{ }\mu\text{Ci }^{137}\text{Cs}/\ell$, and $667\text{ }\mu\text{Ci }^{90}\text{Sr}/\ell$. The composite effluent after passage of 900, 1050, and 1510 CV contained, respectively, 1.52, 3.35, and $86.8\text{ }\mu\text{Ci }^{90}\text{Sr}/\ell$. A small amount of the ^{90}Sr in the effluent may represent complexed strontium since three successive precipitations of inert strontium phosphate only reduced the ^{90}Sr concentration to $3.2\text{ }\mu\text{Ci}/\ell$.

Phase 2 of the capacity run was underway at the end of the quarter. This phase involves passage of the composite (1510 CV) effluent from Phase 1 downflow at 25°C and 4 CV/hr through a fresh 3.0-ml bed of sodium titanate. At report time, a total of 730 CV had passed through the sorbent bed. All the effluent contained $\leq 1.0\text{ }\mu\text{Ci }^{90}\text{Sr}/\ell$. Results of this "hot" capacity test are very satisfactory and encouraging. Hydraulic performance of the titanate powder was excellent throughout. Such excellent flow behavior was realized by following the recommended procedure of washing the titanate

free of fines with 1M NaOH. Strontium removal capacity of the small beds used in this experiment was in reasonable agreement with that expected from the Sandia Laboratories tests.

Hot Cell Pilot Plant Tests - W. W. Schulz

Effort this quarter continued in decontaminating the 222-S Building, Room 1E hot cell in preparation for installation of equipment for scheduled FY 1978 and 1979 flowsheet tests with actual salt cake and residual liquor. Final decontamination of the cell should be completed early next quarter, and it is estimated that the hot-cell setup will be ready for cold shakedown runs by October 1, 1977.

Removal of Technetium-99 - W. W. Schulz

Current and proven technology for removal of ^{99}Tc from salt cake solutions and residual liquors involves sorption of TcO_4^- on a strong base anion exchange resin and subsequent elution with 6M HNO_3 . With this procedure, it is necessary to recover (or destroy) large amounts of HNO_3 during thermal concentration of the technetium product. To overcome this disadvantage, we are examining both solvent extraction and improved ion exchange elution techniques.

Solvent Extraction of $^{99}\text{TcO}_4^-$. Literature⁽²⁻⁴⁾ sources indicate that certain classes of amines and ketones extract pertechnatate ion from alkaline solutions. Accordingly, various reagents were screened (Table IV-3) to test their effectiveness in extracting $^{99}\text{TcO}_4^-$ from simulated salt cake solution. Undiluted Aliquat 336, a quaternary ammonium compound, is a superior technetium extractant; however, the commercially available tertiary amines, Alamine 336 and Adogen 368, are ineffective. Pyridine, a cyclic amine, and cyclohexanone, a cyclic ketone, both extract $^{99}\text{TcO}_4^-$ suitably well from simulated salt cake solutions. Hexone, however,

extracts TcO_4^- effectively only when the aqueous phase contains a small concentration of tetrapropylammonium hydroxide. Tetrapropylammonium hydroxide⁽³⁾ forms an ion association complex with TcO_4^- , which is very soluble in hexone and other organic solvents. The relatively high cost of tetrapropylammonium hydroxide is a potential disadvantage.

TABLE IV-3

CANDIDATE $^{99}\text{TcO}_4^-$ SOLVENT
EXTRACTION REAGENTS

Extractant	D_{Tc} (a)	
	<u>4M</u> NaNO_3 - <u>2M</u> NaOH	<u>5M</u> NaNO_3 - <u>1M</u> NaOH
Aliquat 336 ^(b)	800	970
Pyridine	29.4	12.3
Cyclohexanone	6.08	5.47
Hexone	15.8 ^(c)	
Hexone ^(d)	0.332	
Alamine 336 ^(e)	0.014	0.013
Adogen 368 ^(f)	0.007	0.11

(a) $D_{\text{Tc}} = \text{Conc. } ^{99}\text{Tc organic phase} / \text{Conc. } ^{99}\text{Tc aqueous phase}$

(b) Quaternary amine available from General Mills Company

(c) Aqueous phase contained 0.1M tetrapropylammonium hydroxide (TPAOH)

(d) 6M NaNO_3 -1M NaOH without TPAOH

(e) Tertiary amine; General Mills Company

(f) Tertiary amine; Ashland Chemical Company

Solutions of pyridine (5 to 30 volume percent) in diethylbenzene (DEB) did not extract $^{99}\text{TcO}_4^-$ from 5M NaNO_3 -1M NaOH solution. Extraction of $^{99}\text{TcO}_4^-$ from 5M NaNO_3 -1M NaOH solution by 5-30 volume percent Aliquat 336 - DEB solvents is

currently being investigated. Depending on results of these tests on ease of technetium strippability and on tests with actual waste liquors, undiluted cyclohexanone and diluted Aliquat 336 are the leading candidates for use in solvent extraction removal of ^{99}Tc .

Ion Exchange of TcO_4^- . Our efforts to devise an improved ion exchange procedure for removal of $^{99}\text{TcO}_4^-$ have centered primarily on finding a suitable reagent for low-acid reductive stripping of the sorbed technetium. Of various reducing agents - N_2H_4 , $\text{NH}_2\text{OH}\cdot\text{HNO}_3$, ferrous sulfamate, and ascorbic acid - only 0.1 to 0.5M N_2H_4 solutions appear to be suitable eluents.

On another tack, we are also comparing technetium sorption and elution characteristics of weak base anion exchange resins to those of the conventionally used strong base resins. The hope is to find a weak base resin which will satisfactorily load TcO_4^- from alkaline feeds and from which ^{99}Tc can be suitably eluted with dilute ($\sim 1\text{M}$) HNO_3 . Tests with weak base resins were just underway at report time and no reportable results are yet available. Following completion of batch screening tests, we plan to conduct column tests employing both HNO_3 and N_2H_4 as eluents.

Long-Term Leach Rate Test Program - D. D. Chen

As part of the Criteria Development and Evaluations task, a study was begun this quarter aimed at improving our understanding of leach mechanisms, predictive models, and the reliability of various currently employed leach tests.

A review of the existing leach rate models has been completed, resulting in recommendations for modifying the present Paige leach rate apparatus. Leach testing, using the modified

apparatus, is scheduled to begin next quarter. Hopefully, with control of leachant temperature, pH, and ionic concentration, it will be possible to improve the reproducibility and utility of leach rate data.

Documentation of Conceptual Full-Scale Flowsheet - M. M. Beary

A preliminary flowsheet document (ARH-F-113)⁽⁵⁾ has been issued describing the full-scale process for conversion of Hanford high-level waste to glass. The process considered here involves a chemical separation of the radionuclides from the industrial chemicals, then making glass from the resulting small volume of highly radioactive waste. This version of the glass with radionuclide removal flowsheet is similar to, but contains some conceptual improvements over, the flowsheets used as a basis for the Hanford Defense Waste Document (see Section 4.5 of this report).

Waste Sampler Development Program - N. M. Arnold,
R. C. Lawrence (BNW)

Cold tests were conducted with a modified BNW-developed sampler on simulated salt cake waste material verifying the ability of this tool to retrieve approximately one liter of sample. The material consistency was that of watery oatmeal with particle sizes up to 3/8 inch. Further tests are scheduled to evaluate the entire system and to establish a specific operational procedure.

Cold testing of a prototype surface sampler is in progress. The "Diddler" waste sampler uses a pneumatic cylinder to cyclicly scrape material from the surface layers of the waste tanks, as illustrated in Figure 4-8. Sized for a sample of approximately 300 ml, the sampler is designed to be an inexpensive method of retrieving characterization samples with limited personnel and time requirements.

FIGURE 4-8. The "Diddler" Surface Waste Sampler Concept

These samplers are being developed as backup technology to the ongoing rotary core sampler program.

4.4 STORAGE AND DISPOSAL ALTERNATIVES - R. A. Watrous

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.
- o Identify and develop technology for long-term or final storage of immobilized Hanford high-level waste in near-surface engineered storage facilities.
- o Develop economic and technical criteria against which alternative waste storage concepts may be compared and evaluated.
- o Analyze alternatives and select a preferred storage mode and system design.

Bulk Storage Concept Development - C. A. Knox, B. A. Higley

Concepts for basalt cavern storage of bulk product forms such as cement grout are being investigated. During the report period, a process flowsheet based on the Oak Ridge National Laboratory Grout Process was designed, equipment sized, and product volumes estimated for the conversion of Hanford High-Level Waste to grout. Design of a Basalt Cavern Facility compatible with bulk storage of this grout product was completed, and a cost analysis (relative to other alternative products and storage modes) should be completed early next quarter.

Several variations of this concept are being considered, including a combination of grout and glass in which actinide-bearing sludges are segregated, immobilized, and stored in

the form of containerized glass (allowing actinide retrievability), while nonactinide wastes are disposed of via the bulk storage mode.

4.5 ALTERNATIVES STUDIES AND ASSESSMENTS - R. Y. Lyon

The objectives of this program are:

- o To conduct a comprehensive analysis of all reasonable alternatives being considered for safe long-term management of Hanford defense high-level waste,
- o To prepare technical documentation of the analysis for public review and comment (Hanford Defense High-Level Waste Document),
- o To document for public review and comment the environmental impact of all alternatives being considered (Programmatic Environmental Impact Statement), o
- o To document the environmental impact of the alternative selected for implementation (Project Specific Environmental Impact 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 was rescheduled for completion by April 1, 1977, to permit integration of results into the Defense Waste Document. The analyses will be continued to provide information for the Programmatic Environmental Impact Statement.

Technical Feasibility and Design Studies. During the report period, Kaiser Engineers continued revision and updating of the portion of their report documenting the technical feasibility studies, preliminary design studies, and safety analyses performed. Publication of the final document was postponed until August 31, 1977, to allow time for additional revisions resulting from technical review by Rockwell Hanford Operations personnel in July and August 1977, following final revision of the Hanford Defense Waste Document (HDWD).

Cost Studies. Comparision of estimated costs for similar waste management plans at Hanford and Savannah River indicated that there were differences in several areas, with the Savannah River estimates usually being higher than Hanford estimates. These discrepancies were discussed by ARHCO, Kaiser Engineers, and Savannah River Laboratory personnel in late May. As a result of the discussions, Hanford cost estimates for the radionuclide removal process, the immobilization processes, and several storage concepts were revised upward. However, the Hanford estimates in these areas are still lower than the Savannah River estimates. Cost estimates for the basalt cavern were revised independently to reflect a more austere concept than had been proposed originally.

Risk Analysis. Preparation of risk analysis narrative sections for the HDWD was completed by Nuclear Services Corporation (NSC) in late April 1977. As a result of comments received from review of the HDWD during May, the decision was made to reexamine the accident scenarios, event

frequencies, and radioactivity release fractions; modify them as necessary; and recalculate dose and risk values. A special team of engineers with extensive Hanford experience, was established to make the evaluation. Results of the evaluation will be provided to NSC in early July 1977 for immediate revision of the analysis where necessary.

The decision was made to eliminate the three appendix sections pertaining to risk analysis from the HDWD in order to decrease its size. Nuclear Services Corporation was requested to prepare, by August 31, 1977, a separate reference document containing this information. Revision of the risk analysis and preparation of the reference document are scheduled to start in early July. A revised risk summary (Chapter 6.0) will be prepared first.

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 waste. A draft document was prepared for ERDA review April 30, 1977. Issue of the final document by ERDA is scheduled for September 1977.

A complete draft HDWD was transmitted to ERDA-RL and selected individuals in ARHCO and Battelle-Northwest on May 6, 1977, for review and comment. This was one week later than scheduled. Following incorporation of a few revisions requested by ERDA-RL, the document was transmitted by ERDA-RL to ERDA-HQ, ERDA-ID, and ERDA-SR on May 13, 1977, for review and comment. Comments received by early June were categorized, evaluated, and discussed at a meeting of ERDA, ARHCO, SRL, and Allied Chemical Corporation personnel at ERDA-HQ on June 15 and 16, 1977. The main categories of comment included content of

summary, definition of alternatives, risk analyses, cost estimates, and size, format and readability of the document. Action was initiated to assure response to the comments, in a timely manner, to permit attainment of the major milestone requiring transmittal of the final HDWD to ERDA in late July 1977.

Revisions to be made in cost and risk analysis sections were discussed previously. Redefinition of the major alternatives to provide definite endpoints was initiated. The proposed major alternatives as tentatively defined are shown in Figure 4-9. In addition to the existing "Path A" and "Path B" concepts, "Path C" has been added. This concept is a combination of the other two paths that might be used if retrieval were to be accomplished after a prolonged period of storage in existing facilities. Review of the alternative plans (previously called technical options) was in progress at the end of the report period to determine if additional plans should be added.

Work during the next quarter will include preparation of a final draft document for transmittal to ERDA-HQ by the end of July 1977. Final masters will be revised as necessary and transmitted during August 1977. The document has been assigned the number ERDA-77-44.

Programmatic Environmental Impact Statement (PEIS)

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 waste. 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 PEIS.

FIGURE 4-9. Revised Major Alternatives

Preliminary discussions of format and content were held at the ERDA-HQ meeting on June 15 and 16, 1977. The main effort will be initiated 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 demonstrate technologies required for the selection, establishment, and implementation of a program that provides safe and environmentally sound management of the radionuclides stored in or on the soils and sediments in the 200 Areas' control zone.

Organization of the program is proceeding toward a FY 1978 implementation. The program breakdown structure has been updated to reflect improvements made through an in-depth review. Work packages are being drafted to define FY 1978 activities.

A program plan is being delayed due to the termination of the ARHCO contract and implementation of the Rockwell Hanford Operations contract on July 1, 1977. Reorganization and implementation of a program management matrix throughout the Company is being factored into the plan.

5.1 CONTAMINATED SEDIMENTS CONTROL

Site Characterization - K. R. Fecht and G. V. Last

The activity objectives are: (1) to characterize various intermediate- and low-level 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 Northwest Testing Company received the contract to perform size and calcium carbonate analyses of sediment samples, and has analyzed 500 samples to date. U. S. Testing Company

is performing audits of these analyses. The sediment sample analyses aid in developing geologic maps of the stratigraphy beneath intermediate-level waste disposal facilities.

Currently, the stratigraphy beneath the 216-A waste disposal facilities is being developed for use in the Purex Safety Analysis Report.

The geophysical logging of 240 crib monitoring wells has been completed. The gamma radiation profiles developed from the logs were normalized and reviewed to evaluate the stability of gamma-emitting radionuclides disposed to the ground beneath the crib facilities. Documentation of these data has been completed.

Trench Characterization - S. M. Price, M. K. Additon, and G. L. Wagenaar

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.

Geology and Actinide Distribution. Three 216-Z-1A Trench fringe wells, 299-W18-166, 299-W18-167, and 299-W18-168 (Figure 5-1), were completed during the quarter. A generalized geological profile of the underlying trench sediments,

FIGURE 5-1. 216-Z-1A Trench Fringe Wells

based on logs from these three wells and from previously drilled wells, is presented in Figure 5.2. The depths at which alpha activity was detected with a field "poppy" in Wells 299-W18-166, -167, and -168 are also noted in Figure 5-2. Plutonium and americium analyses of samples from these wells and previously drilled wells (Figure 5.1) are scheduled to be completed by the end of the next quarter.

Actinide Migration. Large volume water samples were collected from Well 299-W18-10, located within the 216-Z-12 Trench, and from Well 299-W15-5, located adjacent to the 216-Z-9 Trench. The samples will be analyzed for actinide content during the next quarter.

Sediment Stabilization - P. J. Wiater and 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.

Subsurface Stabilization - P. J. Wiater

A study was initiated with Battelle-Northwest to investigate biobarriers in the greenhouse. Cheatgrass, Russian Thistle (tumbleweed), and Siberian Wheatgrass were planted in lysimeters with biobarriers placed at a depth of 75 centimeters. The plants were sampled in the middle of the growth season, and will be sampled again at the end of the season. Analysis is for lithium, which is used as a chemical tracer in the experiment.

The S-11 Biobarrier Culvert Plots have been watered to encourage plant growth due to the low available moisture

WELLS²

**FIGURE 5-2. Lithology and Alpha Activity
in 216-Z-1A Trench Wells**

this year. The plants will be analyzed later to determine if breakthrough of any of the barriers has occurred.

A scale model showing the construction of the biobarrier test plots is being prepared to aid in communications.

Surface Stabilization and Cleanup - G. M. Holter

Surface Stabilization. The revegetation field test in the S-17 Borrow Pit area is under observation. Cheatgrass and other grasses have become established in the area, but Russian Thistle predominates. The ground surface remains stabilized, resulting in a drastic reduction of wind erosion and particle resuspension in the area. Observation will be continued to determine if cheatgrass will reseed itself in a chemically stabilized area.

Surface stabilization of the S-16 Trench Biobarrier Plots has also proven effective in reducing erosion. These areas are also under continued observation.

Vegetation growth in the test areas appears to confirm previous laboratory studies which indicated that asphalt emulsion, used as a soil stabilizer, helps to conserve moisture for plant growth.

Vacuum Cleanup Systems. Field testing continues on the mini-vacuum system. Work continues to improve the primary filter system on the vacuum.

Liquid Laundry Effluent Control - G. M. Holter

A BNW document, covering the liquid laundry effluent characterization and treatment studies, has been completed and will be issued upon approval. No further work on liquid effluents is planned at this time.

6.0 ENVIRONMENTAL IMPACT TECHNOLOGY

6.1 HYDROLOGY

Groundwater Hydrology - R. C. Arnett, R. E. Gephart,
R. K. Ledgerwood, W. H. Riggsbee

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

Three unconfined aquifer pumping tests were conducted and completed at Well Sites 699-42-12, 699-57-25 and 699-57-29. This brings to a total of five the number of pumping tests completed this fiscal year. Each test has been undertaken in the Gable Mountain area where very permeable paleochannels underlie the Hanford Reservation. The test results will assist in identifying the flow characteristics of those channels which may form rapid transport paths for radionuclides.

The drilling of four unconfined aquifer pump test sites has begun. Two cable tool drilling rigs are now developing Well Sites 699-40-33 and 699-2-33. Each site will have one 25.4-cm to 30.5-cm (10- to 12-inch) well suitable for long-term high discharge pumping and one to two observation wells. The drilling of all four sites will be completed on schedule in August 1977. Pump testing at each location will begin in FY 1978.

Results of the report ARH-SA-292 were presented at the 1977 Spring Meeting of the American Geophysical Union in Washington, D.C. The report documents the results of two Hanford groundwater scenario studies. The first study examined the hydrologic impact of increased groundwater recharge resulting from

agricultural development in the Cold Creek Valley located west of the Hanford Reservation. The second study involved recovering liquid radioactive waste which was postulated to leak into the groundwater flow system from a hypothetical buried tank containing high-level radioactive waste. The predictive and control capacity of the onsite Hanford modeling technology was used to evaluate both scenarios.

The results of the first study indicated that Cold Creek Valley irrigation is unlikely to cause significant changes in the water table underlying the high-level waste areas or in the movement of radionuclides already in the groundwater. The hypothetical tank leak study showed that an active response (in this case waste recovery) can be modeled and is a possible alternative to passive monitoring of radionuclide movement in the unlikely event that high-level waste is introduced into the groundwater.

Report ARH-C-00017 was issued during the quarter. It is a consultant report documenting the findings of a study to evaluate the potential improvement in accuracy that a three-dimensional flow and contaminant transport model might offer over one- and two-dimensional models such as those currently available. A 414-km² (160-square-mile) areal section of the Hanford unconfined and semiconfined aquifer system was selected for six-year comparison simulations using available data.

Velocity fields calculated by the two-dimensional and three-dimensional models were substantially different. The travel times for tritium particles to reach the river appear to be better characterized by the three-dimensional model based upon comparison with observed results. Surface concentration profiles calculated by the two-dimensional and three-dimensional

models show significant differences over most of the Hanford Reservation. Modeling results can be significantly improved by using a three-dimensional model.

The Tektronix digital cartridge magnetic tape recording unit has been received and installed in the 202-S Building of the 200 West Area. Graphic displays such as the Hanford Reservation background map can now be sent to the Tektronix 4014 terminal at a rate eight times greater than that permitted over the data line from the PDP 11/45 Computer. Data can be entered directly to tape from the terminal keyboard without being connected to the main computer, thus reducing operating costs. Small- to medium-sized programs and data files can now be stored locally.

An interactive computer program to facilitate analysis of large quantities of pump test data has been completed and an informal users' manual received from the subcontractor, Battelle-Northwest. The program is operational on the ARHCO Research Department Tektronix 4014 graphic display terminal.

Offsite Migration - M. W. Grutzeck, P. A. Eddy, R. E. Gephart, and C. M. Ferguson

This activity is aimed at determining the concentration and ultimate fate of any radionuclides that may be present in the uppermost confined aquifers underlying the Hanford Reservation.

Two wells have been cored into the Mabton Interbed. The sites are DB-8, lying just southeast of the 216-B-3 Pond, and DB-9 located between Gable Mountain and Gable Butte. The Mabton Interbed was penetrated at a depth of 297 m (975 feet) below the land surface at DB-8 and 153 m (503 feet) at site DB-9. In both core holes, the interbed consisted of an upper

tuffaceous zone followed by alternating sequences of sands, silts, and clays. During July 1977, a third core site (DB-10) will be completed and then all these sites will have specially designed high-strength well screens installed.

Chemical and radiological data for wells sampled during the first quarter of 1977 (DH-8, DB-4, DB-2, DB-7, and DB-1) have been received. Analysis includes standard water chemistry data as well as radiochemical data for iodine-129, cobalt-60, and ruthenium-106. These data will be detailed in the report "Groundwater Monitoring of the Uppermost Confined Aquifers Beneath the Hanford Reservation".

One interbed was sampled under the new effort of water analysis of aquifers penetrated while core drilling. Radiochemical data for the Rattlesnake Ridge Interbed at Site DB-9 was received. The waters contained 80 pCi tritium/liter and 1.7×10^{-2} pCi iodine-129/liter. Compared to the stratigraphically lower Mabton Interbed, these waters show higher radionuclide concentrations. This probably results from the sampling points' closer proximity to the Gable Mountain and B Pond area where possible cross contamination of the unconfined and uppermost confined aquifers may exist.

In August 1976, a routine analysis of Well DDH-3 resulted in anomalously high concentrations of Cr, Mo, Zn, CO_3^- , Sb, Ba, TOC and dissolved solids. Subsequent well pumping and sampling over the next several months showed a decline in Cr (615 ppb to 160 ppb), Zn (0.7 to 0.025 ppm), CO_3 (107 to 68 mg/liter), Sb (62 to 5 ppb), Ba (64 to 3 ppb), TOC (160-31 mg/liter), and dissolved solids (888 to 678 mg/liter). These data and the observation of an "oil" film on the water are believed to be from external contamination which was probably introduced into the well while drilling was in progress. Molybdenum, however,

underwent an increase over the same period of time. This may have been due to mobilization of molybdenum disulfide used as a pipe joint lubricant. Investigations into these matters and the high tritium content are continuing.

Radionuclide Migration in Groundwater - R. A. Deju,
R. E. Gephart, and G. S. Barney

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

The report, "Three-Dimensional Gravity Investigation of the Hanford Reservation," was received from Dr. B. H. Richard of Wright State University, Dayton, Ohio. It is now under review. The study was designed to construct a three-dimensional geologic model of the Hanford Reservation based upon variation in onsite surface gravity potentials. Residual gravity maps were constructed for the entire Hanford Reservation. All anomalies caused by terrain, unsaturated materials and deep seated geologic variations were eliminated. The residual maps are then maps of the underlying basalt surface and those gravity anomalies created by extensive deposits of coarse, low density glacial outwash. The final report will be documented by September 1977.

Research is underway to determine the distribution coefficients (K_d 's) of selected radionuclides for various sediment types comprising the unconfined aquifer beneath the Hanford Reservation. Strontium distribution coefficients were measured for the Plackett-Burman screening experiments. Thirty-five K_d

values were obtained from experiments using 20 different solution compositions and four different sediments. The K_d values ranged from <0.18 to 10.8 ml/g. The most significant variable was Ca^{++} concentration in the aqueous phase. Calcium ion strongly lowers Sr^{++} sorption. Minor effects on K_d were observed with the following variables: NaNO_3 concentration, sediment source, and $\text{Mg}(\text{NO}_3)_2$ concentration. Only small differences in K_d were observed between using the sand fraction (1.0 to 0.0625 mm) and the silt-clay fraction (<0.0625 mm) of the sediments. Therefore, the sand fraction cannot be ignored in future measurements of K_d .

Distribution coefficients were also obtained for Cs, Ru, and Co. The significant variables which affect K_d values of each of these radionuclides were identified using statistical analysis. Cesium K_d values ranged from 3.36 to >355 ml/g and the significant variables were K^+ concentration, type of sediment (sand or silt-clay), and source of sediment. Ruthenium K_d values ranged from 62.5 to $6,300$ ml/g and the only significant variable was NO_2^- concentration. Cobalt K_d values ranged from 0.11 to $>5,000$ ml/g and the significant variables were NO_2^- concentration, K^+ concentration, and initial pH of the solution. Sediment 3, which has the greatest clay content, was the strongest sorbent for each of the radionuclides.

Long-Term Groundwater Management - W. H. Riggsbee,
R. E. Gephart, and R. C. Arnett

This activity is directed toward identifying potential ways in which high-level waste can come into contact with groundwater, and examining the potential impact to the groundwater regime of past and projected water use activities.

Piezometer installation at thirty-three (33) separate well sites is continuing. To date, piezometers have been completed

at the following locations:

699-S12-29	699-38-65
10-E12	
2-33	50-42
14-38	51-75
20-39	53-55
22-70	55-70
28-40	60-60
31-31	67-51
32-72	96-49

Head permeability tests are being conducted at each site to evaluate the surrounding sediments' permeability as well as to assure satisfactory completion of each structure. Once fully installed, these piezometers will allow water potential measurements and groundwater samples to be taken throughout the entire thickness of the unconfined aquifer.

Geologic characterization of selected core samples from Well Sites 699-43-42, 699-49-100A, 299-W11-26 and 299-W19-10 has been completed. The descriptive parameters determined for the samples included grain size percentages, sediment classification, grain composition, grain shape, sorting, bulk density, particle density and both horizontal and vertical saturated hydraulic conductivity. Results of these analyses are being used to calibrate data obtained through downhole geophysical logs, and for descriptive lithologic and hydrologic maps of the Hanford Reservation's unconfined aquifer.

Order of magnitude errors in the Battelle well contamination computerized data base resulted in incorrect information being presented in Table VI-I in the last quarterly report (ARH-LD-152 B). Examination of the original radiological

analysis sheets revealed that only the tritium concentrations had increased from 1974 to 1976. Total alpha, total beta, cesium-137, cobalt-60 and uranium-238 concentrations show no significant change from 1974 to 1976.

6.2 TRANSPORT

Moisture Transport Studies - R. G. Baca and P. G. Easley

The major objectives of this activity are threefold: (1) to complete the development of a set of mathematical models for the simulation of heat, moisture and contaminant transport in unsaturated sediments, (2) to measure the rates and directions of moisture transport in the Hanford vadose zone within the 200 Areas, and (3) to relate the natural and man-induced moisture transport patterns to the short- and long-term movement of radionuclides in the vadose zone.

During this quarter, principal effort was devoted to program management activities involved in implementing and coordinating the new research program plan developed in the previous quarter. A four-man consultant committee was organized for the purpose of reviewing and evaluating the long-term research plan. In the next quarter, a meeting will be scheduled with the members of the committee to solicit their recommendations. The progress for major program elements is summarized below.

Mathematical Modeling

A generalized computer code is currently under development which will implement the mathematical modeling concepts and techniques previously developed. The computer model is designed to simulate the natural moisture movement in partially saturated sediments as determined by the natural climatic, hydrologic and thermodynamic processes. An initial version of the computer program will be completed this fiscal year as scheduled.

The survey effort was continued to identify and review existing multidimensional models for vadose zone analysis. Work on a planning document which will summarize this survey effort was initiated.

Field Studies of Moisture Movement

Measurement of moisture distribution at the lysimeter site south of 200 East Area has continued. Moisture from precipitation moved down somewhat in April, remained the same in May, and the surface moisture began drying out in June. More moisture was lost from the top two meters of the open-bottom lysimeter than from the closed-bottom lysimeter during June, probably due to differences in surface vegetation. Moisture did not percolate very deeply because of the trace amount of rainfall in April and because the climatic conditions were unusually warm.

The Systems Department developed a computer program to graph the lysimeter profiles as a function of time on a three-dimensional plot. This will facilitate interpretation of the long-term trends observed in the lysimeter data.

All the components for the instrumentation of the old lysimeter site have been ordered. It has been decided to use a solar cell array to power the field instruments.

Two new (ungROUTED) wells suitable for monitoring moisture movement have been drilled north of 241-T Tank Farm. A 12.2-meter (40-foot) deep caisson at the same site is planned for the next quarter. It will be instrumented with psychrometers. These three wells will be monitored over several water years to determine the movement of rainwater into gravelly, coarse sand, as opposed to the fine sand found in the old lysimeters.

The initial moisture profiles of the first two wells were also evaluated to determine if there were significant amounts of water moving laterally to T Farm from the 216-T-4 Ditch.

Preliminary comparisons of the readings in the two wells, one near the ditch, and one away from it, showed no significant differences in moisture content and thus, no apparent water moving laterally from the ditch to the tank farm.

Laboratory Studies

Studies of the permeabilities of sediments at low moisture contents have been continuing, although there have been equipment problems.

Biotic Transport - V. A. Uresk

Identification of biological agents 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

These field studies are designed to provide basic radioecological information about the plants and animals inhabiting the 200 Areas so that their role in offsite transport of radioactive waste can be defined and evaluated.

Review of the first draft of a report summarizing current knowledge concerning ecosystem function, environmental impact and the importance of the biotic component to waste management operations was completed. Preparation of the final copy is in progress and publication is scheduled for August 1977.

Additional plants were excavated near the 216-S-17 Pond site to determine the maximum rooting depth. Cheatgrass was

discovered to root to depths of 180 centimeters (6 feet) while Indian ricegrass was found to root to depths of 200 centimeters (6.5 feet). The feasibility of using cheatgrass for revegetation of waste burial sites is presently under consideration.

Small mammal trapping transects established near Rattlesnake Springs were sampled during April and May to provide control values for a study being conducted at U Pond on radionuclide concentrations and external dose exposures in mice near the pond. Data from the Rattlesnake Springs control area compared very closely with those taken from the control area near U Pond, verifying the validity of the values from the U Pond control area. Gamma exposure to the mice with implanted thermoluminescent dosimeters (TLD) ranged from 1.1 to 1.8 milliroentgens per week at the Rattlesnake Springs location and ranged from 1.2 to 2.3 mR/week at the U Pond control location. The U Pond transects were also trapped yielding 20 animals with implanted TLDs out of a total of 42 captured in April and 24 animals out of a total of 43 captured in May. The dosimeters will be removed and analyzed and the animals will be sacrificed for radiochemical analysis.

The field work portion of a study of Hanford birds of prey has been completed and the data are presently being analyzed for inclusion in one of three reports scheduled for completion this fall. The documents currently planned are: (1) The Behavior and Ecology of Swainson's Hawks; (2) Diurnal Birds of Prey of the Hanford Reservation; and (3) Nocturnal Birds of Prey of the Hanford Reservation.

Analytical results from a two-year study of coots inhabiting 200 Areas waste ponds have been received. The gastrointestinal tract contents continued to show the highest levels of

radioactivity in the animal, followed by the liver, bone and muscle fractions when analyzing tissues from coots collected at U Pond for plutonium-238, plutonium-239, and americium-241. Attempts to define the food source in the pond which contributes to the radioactivity in the gastrointestinal tract continued. The radioactive material appears to be attached to an organic/clay fraction inseparable from the foodstuff or sedimentary material. Radiochemical analyses to determine the cesium-137 levels in the coot gizzard are continuing.

Laboratory Studies of Factors Affecting Plant Uptake of Radionuclides

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.

6.3 SEISMOLOGY AND GEOLOGY

Tank Farm Geology - W. H. Price and P. G. Easley

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.

Geology of the 200 Area Control Zone. Coring of the Ringold formation in 200 West Area was completed this quarter. A generalized north-south geologic cross-section of 200 West Area derived from the core wells is found in Figure 6-1. Based on the coring, a more accurate and detailed paleotopographic map of the basalt surface was also completed (see Figure 6-2).

Two core wells remain to be drilled in 200 East Area.

Hydrological Properties. Two independent laboratories completed measurement of the hydrologic properties for two sets of identical sediments. These data are being compared and correlated to determine if measured differences in permeability can be attributed to packing differences and to see if the conductivity of sediments to strong caustic solutions can be predicted from the hydraulic conductivity for water.

Initial evaluation of the sediment properties measurements indicates that the lithologic classification of sediments is not an adequate predictor of the measured differences in permeability. A computerized multivariate analysis of the physical and hydrologic properties of the sediments has been initiated. It is expected that the results of this analysis will permit better and more efficient sampling and characterization of the sediments in the future.

Mineralogical Properties. Mineralogical analyses of 200 West Area were completed last quarter. Work will begin on the mineralogy of 200 East Area upon completion of the coring.

Research Assistance to Tank Farm Surveillance. Drilling of the 241-T-106 Tank leak has been completed. The twelve

FIGURE 6-1. Generalized N-S Geologic Section Through 200 W Area
(Elevation in Feet above Mean Sea Level)

FIGURE 6-2. Paleotopographic Map of the Basalt Surface Beneath the
 200 W Area (Contours in Feet above Mean Sea Level)

wells that were deepened are currently being perforated and lined with 10-cm (4-inch) well casing prior to grouting.

Analysis of the tank leak data is nearing completion. Final results will be reported in draft form by the end of August 1977.

Seismic Investigations - K. R. Fecht and G. V. Last

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

An immediate response seismic monitoring array, which is part of the plan to minimize releases of radionuclides to the environment due to earthquakes, has four stations, all of which have been calibrated and field tested. Currently, the discriminators, receivers and heliocorders are being installed at the 202-S Building.

A report on the structure and stratigraphy of the Columbia River Basalts on the north flank of the Horse Heaven Hills from Wallula Gap eastward to Van Sycle Canyon is in progress, pending completion of bulk rock analysis at Washington State University.

Trenching across the western extension of the purported Wallula Gap fault has been completed. Geological investigation and logging of the trench walls are in progress. Much interest has been expressed in this site, and several

area geologists have examined the exposures.* The trench is expected to remain open until July 30, 1977 to allow other interested parties a chance to examine the site. A preliminary report on the structure, stratigraphy and history of the site is expected to be completed by the end of the next quarter.

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." The relationships can be used to provide direction and develop criteria for control of toxic substances from routine or episodic transport mechanisms.

Work was initiated on a paper equating hazards available to risk assessment. The paper will give examples of hazards to man from given waste site types, conservatively assuming worst case frequency of episodic events. The hazard index, transport factors, and hazards available over a 10^8 -year period will be plotted for different distances from the waste site. Recommendations for needed programs such as Maximum Permissible Concentration (MPC) studies and Transport Factor studies will be summarized.

A request to submit a proposal to the Environmental Protection Agency, Proposal (RFP) WA77-B096, was received. It concerns toxicological test methods used in estimating potential human health hazards. The request will be reviewed, and if

*Messrs. Donald Swanson and Richard Waitt, U.S. Geological Survey; Mr. Bill Kiel, Washington Public Power Supply System; Mr. Robert Deacon, Shannon and Wilson Consultants; and Messrs. Jamie Gardner and Fraser Goff, ARHCO consultants.

a proposal is not made or can not be made, an appropriate letter showing our interest and willingness to assist the winning bidder will be sent.

6.5 SPECIAL STUDIES

Gaseous Waste Monitoring Study, Battelle-Northwest

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

The BNW study is 99 percent complete. Evaluation and documentation of findings for ten stack monitoring systems have been completed. Documentation of experimental findings at the 291-Z-1 and 296-Z-3 Stacks is about 50 percent completed. When this document is received, the BNW study will be completed.

Results of BNW's evaluation will be used to upgrade existing gaseous waste monitoring systems.

7.0 ANALYTICAL METHODS TECHNOLOGY DEVELOPMENT

The objective of the analytical methods technology mission is to provide the integrated development of instrumentation and technology to measure the concentration of radioactive and nonradioactive constituents in samples from Production, Separation, Waste Management, Environmental Monitoring, and Research. The developed technology will be transferred to the Analytical Laboratories and is designed to improve the quality and/or quantity of support afforded to processing component.

7.1 ANALYTICAL INSTRUMENTATION DEVELOPMENT

The objectives of this activity are to develop cost-effective analytical instrumentation to provide routine support for ongoing programs and to reduce radiological exposure and the potential for safety hazards by reducing the amount of sample manipulations required.

Californium Multiplier (CFX) - Subcritical Assembly -

V. W. Hall, J. K. Hartwell, and R. R. Rietz

The status of work on the CFX system by the IRT Corporation is summarized:

<u>Component</u>	<u>Percent Completed</u>
Mechanical Design and Drafting	99+
Mechanical Fabrication	99
Electrical Design and Drafting	99+
Electrical Fabrication	99
System Testing	99
Ancillary Equipment	90
Operations Manual	25

The CFX assembly and the delayed neutron counter (DNC) have been received at Hanford. Inspection of the components has not been completed.

Documentation of the system, including an operating manual, has not yet been provided. Blueprints of the assembly and a map of the fuel plate positions have been obtained, although some are still classified proprietary by IRT and as such are not available for unlimited distribution.

Tests of the CFX unit in the IRT facilities indicated that, when scaled up to the 40 mg ^{252}Cf to be used at Hanford, the neutron flux below 1.1 eV will be 9.22×10^9 neutrons/cm².

The k_{eff} of the system will be 0.9896.

At Hanford, a seismic evaluation of the containment vessel, a hazards review, and an independent criticality safety review are in progress. The Safety Analysis Report (SAR) is being written as information becomes available. Construction of ventilation modifications, delayed neutron counter tank installation and pneumatic transfer system installation has not yet been started.

Alpha Counting Instrumentation - J. K. Hartwell,
D. E. Griffing, N. L. Row

An alpha energy analysis (AEA) data reduction software package has been written to analyze pulse height spectra obtained from eight alpha detectors. The hardware instrumentation for the alpha counting system has been operational for the past quarter but in a manual mode.

The AEA data reduction software will function to identify isotopes present in each alpha spectra, to quantify the amounts of each isotope, to approximate the uncertainty associated with each quantity calculation, and to generate a hard copy report without manual manipulation of the instruments or the raw data.

Presently the AEA software package is awaiting acceptance by the Chemical Sciences Section of the Research Department and the Analytical Laboratories. Official test results of over 150 NBS and laboratory generated AEA spectra were transmitted to the Chemical Sciences Section on June 24, 1977 as the first step toward code acceptance. The accuracy of the results may be improved by further adjustment and calibration of parameters in the AEA software.

The present version of the program was written using structured design and programming techniques resulting in a highly modular, easily modifiable, well documented, and reliable code. Approximately, 70 percent of the previously reported AEA program was completely rewritten to obtain the current program. Design inconsistencies, algorithm deficiencies, and poor programming techniques necessitated such an extensive rewrite.

This work will complete a Research and Engineering five-year plan milestone.

Aerosol Monitoring - Air Sample Counting - J. K. Hartwell

A revised system to monitor radionuclide content on particulates in room air has been initiated. This program is about 1 1/2 months behind the schedule established in the five-year technology plan. This is due to reestablished priorities of other programs.

An internal document for estimating detectable limits for the proposed laboratory air sample analysis method has been prepared. Using the detectable limit data documented therein, purchase specifications have been prepared for the components of an aerosol counter. This system is being designed to monitor levels to one tenth the maximum permissible concentration expressed in the Energy Research and Development Administration Manual 0524, Table II. This system will be largely automatic to facilitate processing the large number of samples generated to provide legal documentation of air quality.

Inductively Coupled Argon Plasma Spectrometer - C. S. Homi and P. A. Meyer

An inductively coupled argon plasma spectrometer (ICP), a multielement measurement instrument, has been received and set up. Checkout tests have been completed by both Applied Research Laboratory, the manufacturer, and by Research Department personnel. These tests indicate that the unit meets or exceeds the sensitivity, stability, and precision specifications for 29 of the 30 elements requested. These elements are quantified simultaneously.

The unit is currently used to provide analysis on non-radioactive samples as a development effort to identify anomalies of the system. The off-gas scrubber system has been designed and is undergoing review for fabrication. When the off-gas system is installed, the ICP will be used routinely for process support and effluent monitoring samples which require multielement analyses. This will supplement atomic absorption spectrometric methods while reducing sample preparation and analysis time. This will also have the effect of reducing personnel exposure to radioactive materials.

Problem areas which have been identified while analyzing nonroutine samples include:

- o Dissolved solids greater than two percent restrict the flow of sample through the nebulizer to the excitation source, an argon plasma. This restriction increases background response and therefore makes the instrument less sensitive. This may have deleterious effects on some trace element analyses. Development will continue to overcome this limitation.
- o Light scattering and interference tests on several elements have been conducted. Light scattering from calcium becomes noticeable on adjacent channels when the concentration reaches $\geq 20 \mu\text{g/ml}$. Efforts are continuing to reduce light scattering. Direct line interferences have been identified for six combinations of elements. These can be mathematically corrected and remain analytically correct.

An imbedded microprocessor and software package on loan from the manufacturer is currently used to control the instrument and manipulate data. The permanent installation will be controlled by the in-house analytical mini-computer system. This development will soon be initiated. The additional storage and flexibility afforded by the larger computing system is expected to result in significant time and calibration savings.

Investigations of the instrument capabilities will continue on hydrological, dissolved basalt, synthetic waste management and research program samples.

Gas and Liquid Chromatography - S. J. Johnson

The capabilities of the recently installed Hewlett-Packard model 5840 gas chromatograph have been extended by installation of an inlet splitter and a 60-meter glass capillary column. Performance tests of the column indicate the column has 2.7×10^4 theoretical plates whereas the calculated value showed $> 6 \times 10^4$ theoretical plates. The reason for the disagreement is being investigated as time permits. The improved system (compared to a stainless steel-packed analytical column) has produced highly resolved chromatograms. This has made possible monitoring impurity concentrations of 20 major constituents of dihexyl-N, N-diethylcarbamy-methylene-phosphonate (DHDECMP) during pilot plant tests of cleanup facilities and procedures. Analyses of welding gases have been provided on a development basis. The technique was highly successful in determining off-standard gas mixtures.

A fluorescence detector has been received for installation on the Hewlett-Packard model 1080 high performance liquid chromatograph. This will make possible analyses of organophosphates and supplement the existing ultraviolet detector now in use.

Implementation of this liquid chromatograph is proceeding on schedule. Standard solutions of DHDECMP and impurity constituents normally found are not available commercially. This instrument has proved useful in preparing these fractions for gas chromatographic evaluation.

Irregularities in pump pressures and solvent flow rendered the instrument inoperative until factory representatives were able to resolve and repair the malfunctions. Continued operation will determine if these are to be recurring problems.

Development activities will continue to determine strontium complexing power, especially for ethylene-diamine-tetraacetic acid and hydroxy-ethylene-diamine-triacetic acid (EDTA and HEDTA), using reverse phase liquid chromatography. This will provide needed results for the radionuclide removal development efforts of the Research Department.

An ion exchange liquid chromatograph has been received and is undergoing performance tests. This unit is expected to provide routine analyses of high salt samples for low concentrations of sulfate, nitrate, nitrite, phosphate and chloride ions. If successful, this instrument will require little attention and therefore will reduce radiation exposure while providing high sample throughput. Development work is ongoing. This item was not included in the five year technology plan but appears to satisfy a high priority need which has developed.

Total Organic Carbon Analyzer - A. G. Miller

The cumulative deficiencies in the capability of the presently used carbon analyzer to effectively determine organic carbon in Hanford nuclear waste materials forces a change to an improved method. The object of this subtask is to provide instrumentation capable of providing reliable and rapid determinations of total organic carbon (TOC) in Hanford nuclear waste samples. The effort has been three pronged:

- o To evaluate the ability of the presently used carbon analyzer furnace to adequately combust samples, with or without a slight modification.
- o To investigate acquiring a different combustion furnace to be used in conjunction with the presently used carbon analyzer detector/readout instrumentation (which performs satisfactorily).

- o To survey the capability of all total organic carbon analyzers on the market for their ability to perform satisfactorily on Hanford samples.

Tests have been conducted with synthetic waste samples. The results of these tests indicate that for the high salt waste of concern:

- o The presently used combustion furnace is not capable of providing high enough temperatures to combust the complexants of interest.
- o An incompletely burned sample was the source of the memory effects previously experienced.
- o The presently used instrument does not adequately control the current to the sample filament. Over-current frequently occurs.
- o A different combustion furnace is necessary.

Three combustion furnaces are being investigated for use in conjunction with our CO₂ detector. They are the TOC furnaces sold by Beckman Instruments and Coulometrics Incorporated, and a laboratory tube furnace for which a suitable combustion tube would be designed. All three would operate at a constant temperature of 900° to 1000° C, eliminating the cycling of the Oceanography furnace.

Our design of a combustion furnace will ideally be for a very small tube furnace which would have to be purchased. Provisionally, a combustion tube may be produced for a larger tube furnace which was used for carbon determinations in the past.

A Beckman TOC instrument owned by Teledyne Wachang, Albany, Oregon, will be evaluated for performance on Hanford synthetic waste liquor containing the above described complexants. A similar investigation of the Coulometrics equipment will wait for the results of the in-house furnace design.

Atomic Absorption Spectroscopy - S. G. Metcalf

The Varian atomic absorption spectrometer has been successfully interfaced through CAMAC to the NOVA mini computers. The interface allows automated data acquisition and reduction for a series of experiments designed to quantitatively evaluate the instrumental parameters limiting the precision of atomic flame emission spectroscopy.

In conjunction with the analytical automation team, the CAMAC system has been extended and interfaced with the atomic absorption spectrometer. The analog recorder output is input to a voltage-to-frequency converter (VFC) through a buffer amplifier. The output of VFC enters the CAMAC system through a dual scaler-timer to provide an integrated digital signal. This results in a 100-fold improvement in the instruments quantization noise (i.e., two additional significant figures were added to the instruments' readout). This is most important at very low light levels, i.e., low transmittance in the atomic absorption mode, or low concentrations in the atomic flame emission mode. The gain of the photomultiplier tube on the instrument as a function of bias voltage has been calibrated to allow the accurate determination of quantum noise (shot noise) and to evaluate the general performance of the tube.

The program allows for a variable number of signal measurements with variable integration time. Usually 30, one-second

integrations are done. They are averaged, and the standard deviation is calculated. It is assumed that any value more than three standard deviations away from the mean is a "flier". Using this assumption, the data are tested and, if appropriate, the data are rejected and an error message is returned to the operator. It has been observed that 10 to 15 percent of the data during day shift is rejected due to high fliers and during hours of shift change, 50 to 60 percent of the data show high fliers. A reasonable cause would be voltage spikes in the building power. Consideration should be given to this possibility by other instrument users with respect to their own instruments' stability. Next, the background signal and noise are subtracted from the total signal and noise, both of which are input variables experimentally measured earlier, to give the analyte emission signal and the total analyte emission noise. The total quantum noise (shot noise) is calculated and subtracted from the total analyte emission noise to give the analyte emission flicker noise. Flicker noises are of special interest in that they are due to instrumental aberrations and may be reduced by improving the instrument. An example is varying burner efficiency. The relative standard deviation (flicker factor) of the analyte emission signal is calculated next. Lastly, the signal-to-noise ratio, which is the reciprocal of the relative standard deviation in the total analytical signal, is calculated. The results of these calculations are output at the terminal next to the instrument and on command output on the line printer in tabular form.

A series of experiments has been designed that will allow the determination of the optimum value for each instrumental parameter and will identify the factor that most limits the instruments' analytical performance.

7.2 ANALYTICAL METHODS DEVELOPMENT

Analytical methods and technology are developed within the scope of this task and transferred to the Analytical Laboratories for support to production, research, and quality assurance programs. The activities are designed to produce well investigated and documented analytical methods.

Iodine-129 Measurements - C. B. Honaker

A system to measure the iodine-129 concentration in the Purex off-gas continues. There are several facets to this project which are being investigated independently. They are:

- o Release of the iodine-129, whether inorganic or organic salt, then trapping the released iodine. In the procedure currently used to separate iodine from a sample, the iodine is converted to elementary iodine, distilled and trapped as an iodide in 10 percent sodium bisulfite solution. The iodide is converted back to iodine, extracted into toluene and combined into a liquid scintillation counting cocktail.
- o Quantification of iodine-129 collected from a sample. There are several measurement techniques possible. Integrating the separation phase and the quantification phase has caused severe difficulty. The bisulfite solution containing the iodine was passed through an ion exchange paper; however, this produced a five-fold decrease in the amount of iodide retained by the paper. Hydroxylamine hydrochloride was tested with an approximate ten-fold decrease in trapped iodide. Hydrazine is currently being tested. If this does not provide adequate characteristics, precipitation techniques will be tested. Adsorption on ion exchange paper is attractive for quantitative measurements by intrinsic germanium detection

of the X-ray emission. This technique is specific for iodine without interference from fission product activity. Liquid scintillation counting is another alternative for detecting beta activity but is not necessarily specific. Liquid scintillation counting will be required for carbon-14, krypton-85 and tritium; all are required analyses for Purex startup.

Rare Earth Measurements - C. B. Honaker

An improved rare earth separation technique is being investigated on a high pressure ion exchange column (3 mm by 50 cm) bed of 200 to 400 mesh resin. After loading, the rare earths are eluted with α -hydroxybutyric acid, to which 4-(2-pyridylazo) resorcinol is added. This chromophone is added and mixed in an in-line chamber then pumped through a spectrophotometric cell for concentration measurement. Preliminary tests appear capable of providing the necessary precision and accuracy.

Particle Size Distribution - A. G. Miller

The effort to define the parameters to measure the particle size distribution of water soluble crystals, primarily sodium nitrate, which form in concentrated nuclear waste liquids is complete. Improved techniques now permit determination of particle size distribution on samples of very small crystals (within the limitations of the HIAC particle counter).

The slurry received is filtered, and the solids fraction is contacted with successive aliquots of acetone to remove the interstitial mother liquor. The dry solids are suspended in nitroethane as discrete crystals by stirring and ultrasonic vibration to break up agglomerates. The crystals are screened to particle size ranges compatible with the particle size

analyzer. Nitroethane is used as the dispersing medium during measurement.

An evaporator-crystallizer pilot plant sample, which contains solids of a very fine particle size, has been used as a worst case sample to appraise the effectiveness of the analytical techniques developed. A large fraction, greater than 85 percent of the particles, is less than 10 μm in diameter. Sixty percent by weight of the crystals is less than 50 μm . The multitude of very fine crystals makes it difficult to effectively dry this sample for determination of particle size distribution. Careful application of the procedure, especially recent improvements, has accomplished the particle size distribution determination of this sample.

Waste Characterization - A. G. Miller

A limited investigation has been performed on the insoluble fraction remaining after 6M HCl dissolution of waste tank sludge. Laser-Raman and X-ray emission and diffraction spectroscopic techniques were used in the identification.

The fraction consisted of a grey powder and deep rose colored crystals, 1 to 2 mm in diameter. Species identified in the powder include: Gibbsite - $\text{Al}(\text{OH})_3$, Boehmite - $\gamma\text{Al}_2\text{O}_3 \cdot \text{H}_2\text{O}$ and γAlOOH , $\text{Al}_{4.5}\text{O}_{4.5}(\text{OH})_{4.5}\text{Cl}$ (approximate composition); minor components include: Akatoreite - $\text{Mn}_9(\text{Si},\text{Al})_{10}\text{O}_{23}(\text{OH})_9$, Natrolite - $\text{Na}_2\text{Al}_2\text{Si}_3\text{O}_{10} \cdot 2\text{H}_2\text{O}$, $\text{Fe}_2(\text{TiO}_4)_3$, Kirschsteinite - CaFeSiO_4 , Iscorite - $\text{Fe}_7\text{SiO}_{10}$, and Zeolite K-1 - $\text{K}_2\text{Al}_2\text{Si}_2\text{O}_8 \cdot 3 \cdot 8\text{H}_2\text{O}$. The crystals were identified as Almandite - $\text{Fe}_2\text{Al}_3(\text{SiO}_4)_3$ or $3\text{FeO} \cdot \text{Al}_2\text{O}_3 \cdot 3\text{SiO}_2$.

Actinide Analytical Program - S. J. Johnson

Detailed procedures for quantifying concentrations of plutonium, americium, neptunium, curium, thorium and uranium in caustic, high salt waste solutions have been provided to the Analytical Laboratories for routine use. These procedures include

sample preparation and solvent extraction techniques specific for separation and concentration of each of the constituents of interest. These procedures also provide operating data on the electrodeposition technique and equipment necessary to prepare radioassay discs. Plutonium, neptunium, and americium concentrations are quantified by alpha spectroscopy, corrected by spike recovery, while thorium and uranium concentrations are measured spectrophotometrically. The procedures provide data reduction, reagents and equipment required. Final documentation of the methods will be completed by September 1977.

7.3 ANALYTICAL AUTOMATION - D. L. Anderson, W. F. Bauer, and K. G. Braithwaite

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

The necessary hardware to provide data acquisition has been configured and interfaced to the Varian Atomic Absorption Spectrometer for the analysis of radioactive samples. Computer programs have subsequently been developed to evaluate the CAMAC hardware that will be incorporated into the finished system. Data from actual standard solutions run on the spectrometer have been obtained directly by the computer via the test programs with excellent results.

Efforts are continuing on the automation of the laser-Raman spectrometer. An interim version of the final system is nearing completion and implementation for initial user evaluation should begin soon. Features included in this version will be:

- (1) Enhanced data acquisition capability including:
 - o Multiple methods of scanning (i.e., constant signal/noise ratio, constant standard deviation, or constant scan rate).

- o Rapid scan or quick survey of spectral regions of interest with a printout of digital spectral intensity for any given point in the spectra.
- (2) Data treatment including:
- o A nine-point data smoothing routine.
 - o Full on-line plotting capability of all or any selected region of a given digitized spectrum.
 - o On-line output to the remote TTY of any selected region of a digitized spectrum.
- (3) Data reduction including:
- o A program for the data acquisition and direct determination of nitrate.
 - o Calculation routines for final calculation of each analyte from user input analyte intensities.
 - o Generation of an archive file of final results.

Final plans are being formulated and work is beginning on the automation of Alpha/Beta counting instruments. These instruments are used primarily for the analysis of environmental samples and the analysis of air samples. Because of the importance of the air sample program, instrument downtime, that would be involved in the development and testing of the interface hardware and computer programs, would not be tolerable. Therefore, initial efforts will be directed toward the automation of the environmental counter. This will allow for the development and testing of the hardware and data acquisition programs that will ultimately be employed in the automation of the air sample counting. An interface for the counting room Beckman Wide Beta counter, which will enable data to be acquired by the computer via CAMAC, has been designed and is being fabricated. This data will subsequently be placed in a disk file and accessed by data reduction routines.

Planning meetings have been held to discuss hardware configurations and data reduction capabilities required for the automation of the Inductively Coupled Argon Plasma Spectrometer. The instrument has been installed and is being operated on a temporary basis interfaced to a vendor-owned PDP/11-05 minicomputer to handle the data acquisition and data reduction. The instrument was purchased, however, with the intent to interface it to the NOVA computer system via CAMAC.

A software bug related to the use of a file access technique was encountered in the Data General Real-Time Operating System (RDOS). The existence of this bug would not allow the use of a multiple file accessing technique used in the counting room automation programs. Data General personnel have supplied a temporary fix until a permanent change to the operating system has been implemented in their next software revision.

Extensive hardware installations have been made to the NOVA computer system including:

- o An expansion chassis for the NOVA 840 computer.
- o A Hardware Floating Point unit for each of the NOVA 830 and NOVA 840 computers.
- o A Synchronous Line Adapter for the NOVA 840 computer.
- o A Dual Diskette Disk Drive.

This effort has caused a moderate amount of unavoidable scheduled and unscheduled disruption of routine service. Extensive efforts were directed toward restoring the system to normal and are now complete.

Research Assistance to 234-5 Building

A full core of sludge was obtained from the 361-Z Tank using the suction-type tool as modified during the previous quarter. The operation indicated that the modified tool can be used to obtain a representative sample of alpha-contaminated, semi-fluid material. Documentation of the sampling method is scheduled to be completed by the end of the fiscal year.

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 these data, along with the instrumentation, will provide a basis for the long-term surveillance of these sites.

Beta Assayer - L. E. Bruns

The weight of the cart assembly has been reduced about 34 kilograms (75 pounds) and the center of gravity has been shifted to ease the mobility in desert terrain. The manual is about 95 percent complete and the unit will be ready for 200 Area field testing in the fourth quarter of fiscal year 1977.

Dev-Van-I - L. E. Bruns

A new Onan generator was received and has been installed. All detector systems were checked and calibrated. One system was loaned to ARHCO's Equipment Development and Instrumentation Section for inventorying ARHCO's plutonium. The surface Ge(Li) unit collimator was redesigned and is being fabricated. A Si(Li) detector was purchased along with electronic support equipment that made it compatible

with existing Dev-Van-I electronics and suitable for obtaining in-field plutonium analyses. The detector system was developed on a cooperative program with National Nuclear Corporation, Nuclear Equipment Corporation and ARHCO. It will detect samples of plutonium from 0.02 nanocurie plutonium per gram to milligrams plutonium per gram. It will be used for the Actinide Trench program. The Dev-Van-I auxiliary battery system was checked for operability and safety. For minimizing hydrogen buildup, a good air dilution will be assured.

Dev-Van-II - L. E. Bruns

Dev-Van-II was received and was inspected at Gearhart-Owens in Fort Worth, Texas. A few minor changes were promptly made and the van was driven from Texas to Richland by Transportation, Utilities, Maintenance, Services, Division personnel. Dev-Van-II has two Onan 5-kilowatt generators to operate air conditioning or heater systems and all electronic equipment. The draw works is operated by the truck motor; everything can be operated from the seat in the van, i.e., generators, truck motor, draw works, electronics, etc. The current accuracy of the depth is to 0.1 meter, however, one can interpolate to 0.05 meters (2 inches). Another system is being studied that will allow a 1-centimeter accuracy.

The unit is now in the 300 Area and Battelle-Northwest is testing the detector system; it has been stored for over two years. The storage evidently caused crystallization in place and, on movement of the downhole systems, O-ring leaks developed. One unit, the intrinsic germanium (IG), has already been sent back to Princeton Gamma Tech for repairs. The second unit, an 80 cubic centimeter Ge(Li)

system, will be used to test the electronics and, when the IG is returned, the Ge(Li) system will be sent back for replacement by a large IG which is more efficient and does not demand liquid nitrogen at all times.

There are available neutron and sodium iodide gamma counters that can be used for initial downhole tests while the high-resolution detectors discussed above are being repaired.

The final details on the downhole directional system have been completed. With the changes, no test at Hanford is required.

"Passive" Activation System - L. E. Bruns

Tests to date in a small calibration facility in the 300 Area have shown the following:

$^{63}\text{Cu}(n,\gamma)^{64}\text{Cu}$	Limit of 4.6 ng Pu/cc soil (Pu as fluoride).
$^{63}\text{Cu}[\text{Cd}](n,\gamma)^{64}\text{Cu}$	Limit of 56 ng Pu/cc soil (Pu as fluoride).
$^{27}\text{Al}(n,p)^{27}\text{Mg}$	Shows possibilities of distinguishing fluoride from oxide.
$^{24}\text{Mg}(n,p)^{24}\text{Na}$	Isotopically selective and not subject to different chemical compounds; limit of 6.6 $\mu\text{g } ^{240}\text{Pu/cc}$ (as fluoride).
$^{115}\text{In}(n,\gamma)^{116\text{m}}\text{In}$	Measure total thermal and resonance flux, sensitivity 14 ng Pu/cc (as fluoride).
$^{115}\text{In}[\text{Cd}](n,\gamma)^{115\text{m}}\text{In}$	Measure resonance portion of neutron spectrum, sensitivity 22 ng Pu/cc (as fluoride).

A breakthrough was realized this quarter with the indium foils. The indium activation product has only a 56-minute half-life, yet the cross-section for neutrons was sufficient to measure concentrations of plutonium as low as 14 ng Pu/cc as the fluoride. Also, a less sophisticated counting system (sodium iodide-thallium) mounted in Dev-Van-II could give plutonium and americium values in the nanocurie to microcurie range. With a gamma reading for americium, both plutonium and americium values could be obtained downhole.

The ARHCO-financed program with Battelle-Northwest in the 300 Area was stopped due to lack of funds. Important experiments recommended for further study in the 300 Area are: (1) determine effects of moisture; (2) determine methods of approximating plutonium isotopic composition in-situ; and (3) determine methods of estimating plutonium species, e.g., fluoride versus oxide.

Dev-Van-III - L. E. Bruns

Two proposals were received on development work for Dev-Van-III, one from Scientific Applications, Incorporated and one from Intelcom Radiation Technology. These studies involve methods of determining soil and sediment density, porosity, moisture, elemental composition, plutonium and uranium compositions. Studies are also aimed at determining the above in high gamma fields such as high-level waste salt cake. Studies need to be made on a continuum of energy sources like californium versus a monoenergetic source of neutrons. Various companies were contacted about building the van.

Low-Level Drum Counter - L. E. Bruns

One proposal was received verbally on initial development studies for measuring plutonium in 208-liter (55-gallon) drums at the 10 nCi Pu/g level regardless of the density. The conceptual unit would consist of a monoenergetic neutron flux bombardment and prompt neutron interrogation.

Calibration Facility - L. E. Bruns

The four-quadrant system was completed during the quarter. Considerable investigation was made on the optimum matrices to use with the final decision on matrix and sampling as follows:

Quadrant 11	Delmonte Sand - 10 one-half liter random samples
Quadrant 22	Typical 200 Area Sediments - 10 nineteen-liter buckets taken at random
Quadrant 33	Synthetic Salt Cake
Fine - Sampled at 735 and 870 kg
Medium - Sampled at 980 and 1,300 kg
Coarse - Sampled at 850 and 1,079 kg
Lumps - Sampled at 829 and 847 kg
Quadrant 44	Solid Waste Burial Simulation (one 19 liter bucket sample).

The laboratory will do density and moisture determinations on samples from all quadrants to verify field bulk densities and estimated moisture determinations.

The Delmonte sand gives a homogenous matrix which can be used as a basis for matrix standardization. Its uniform elemental composition will be important in calibration of soil types. The 200 Area typical sediments will be variant

with depth, and the extent of this variation and its effect on gamma and neutron readings will be determined. The typical Hanford burial ground contains shredded boxes, plastic, gloves, metal and other materials. Alternate layers of typical sediments and the sediments and the simulated solid wastes constitute this quadrant. The salt cake is a variation of fine wet salt cake to hard and dense lumps of salt cake.

Initial experiments planned for the fourth quarter 1977 include attenuation values for gammas and neutrons in the different matrices, moisture determinations by an americium-beryllium instrument, and comparison of 4π gamma readings with point and line source readings.

The calibration facility now consists of a 4π gamma standard and the four quadrant system. The remaining unit in the 4π plutonium standard will consist of two layers, each being 1.2 meters deep and 1.2 meters in diameter. One layer will be at about 100 nanocuries Pu/g, the other at 10 nanocuries Pu/g.

8.0 ENVIRONMENTAL CONTROL TECHNOLOGY

8.1 MANAGEMENT OF SURPLUS FACILITIES AND EQUIPMENT

Disposition Methods Research and Development

Contaminated Equipment Volume Reduction (CEVR) -

G. A. Beitel

(D. A. Turner, Program Manager)

This program proposes to demonstrate that multiton, 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 as the size-reduction tool, and a cold wall vacuum furnace for melting. Cold testing of volume-reduction equipment will begin by the end of FY 1978. Hot testing of volume reduction equipment will begin in FY 1982. Melting of small samples (1 kg) of metals contaminated with both trans-uranics and fission products is planned to study the redistribution of radionuclides. This small scale testing is separate from the proposed prototype melter.

The Purex dissolver decontamination study is scheduled to be reactivated the first week of July. During this quarter, the standard operating procedure to cover the decontamination effort was written.

Requests for bids to build the cold wall vacuum furnace were sent out and were due the last week of June. Only two positive responses were received, one of which included a two-week extension request which was granted. This will delay contract signing, and limit FY 1977 expenditures. As a result of the inability to get a contract out in June 1977 as initially hoped, it appears that the program budget will be underrun by about \$100,000. These delays can be attributed in part to the approval and contracting procedures required.

Rebids for the 15,000 ampere power supply required for both the arc saw and vacuum furnace programs were rejected and the specification was rewritten to clarify some ambiguous sections. Rebids are due July 14, 1977 and a contract should be awarded shortly thereafter.

The laboratory scale contaminated melt task is preparing to equip a laboratory in which the tests will be conducted. A tilt-pour induction furnace will be purchased. Bids have been received, but complications exist which have prevented placing the order. A glovebox has been obtained from surplus, free of charge, and the laboratory space is being cleaned up (at a cost of \$80,000) on another program. A \$25,000 work order on this program has been written to provide safety windows and installation of the glovebox.

REFERENCES

1. ARH-SA-285, March 1977, M. J. Kupfer and W. W. Schulz, "Fixation of Hanford Sludge by Conversion to Glass."
2. G. E. Boyd and Q. V. Larson, J. Phys. Chem., 64, 988 (1960).
3. J. B. Gerlit, Proc. 1st Internat. Conf. Peaceful Uses of Atomic Energy, Geneva, 1955, Vol. 7, p. 145, United Nations, New York, 1956.
4. R. E. Foster, W. J. Maeck, and J. E. Rein, Anal. Chem., 39, 563 (1967).
5. ARH-F-113, June 1977, M. M. Beary, L. A. Chick, P. C. Ely, S. A. Gott, "Preliminary Flowsheet for Conversion of Hanford High-Level Waste to Glass."

DISTRIBUTION

Energy Research and Development AdministrationRichland Operations Office

J. C. Cummings
O. J. Elgert (3)
D. R. Elle
R. E. Gerton
R. B. Goranson
G. J. Miskho
M. W. Tiernan

Albuquerque Operations Office

D. Davis, Jr.

Headquarters

R. G. Bradley
R. B. Chitwood
C. R. Cooley
G. W. Cunningham
G. M. Daly
C. H. George
O. P. Gormley
C. A. Heath
G. K. Oertel
A. F. Perge
G. B. Pleat
R. W. Ramsey, Jr.
D. C. Vieth
R. D. Walton

Idaho Operations Office

J. P. Hamric

Nevada Operations Office

R. R. Loux

Oak Ridge Operations Office

V. J. D'Amico
E. H. Hardison
J. J. Schreiber

DISTRIBUTION (continued)

Savannah River Operations Office

E. S. Goldberg

N. Stetson

Technical Information Center (2)Nuclear Regulatory Commission, Headquarters NS

R. Cooperstein

Allied Chemical Corporation

A. P. Roeh

C. M. Slansky

Argonne National Laboratory

M. J. Steindler

Battelle Pacific Northwest Laboratories

N. E. Carter

D. B. Cearlock

G. W. Dawson

R. L. Dillon

J. R. Eliason

J. W. Finnigan

E. R. Irish

J. H. Jarrett

H. V. Larson

R. P. Marshall

J. L. McElroy

J. M. Nielson

R. E. Nightingale

D. E. Oleson

L. T. Pederson

A. M. Platt (2)

J. A. Powell

W. H. Rickard/R. E. Wildung

L. C. Schwendiman (2)

C. L. Simpson

C. M. Unruh

H. H. Van Tuyl

B. E. Vaughan/R. F. Foster

DISTRIBUTION (continued)

Center for Water Resources Research, Desert Research Institute

P. R. Fenske

E. I. du Pont de Nemours and CompanySavannah River Laboratory

J. L. Crandall
C. B. Goodlett
H. T. Groh/A. S. Jennings
C. H. Ice
J. A. Kelley
B. C. Rusche/D. A. Orth

Savannah River Plant

E. O. Kiger
S. Mirshak
W. J. Mottel

Wilmington, Delaware

R. E. Naylor
J. F. Procter

Holifield National Laboratory

J. O. Blomeke

Los Alamos Scientific Laboratory

F. A. Guevara

Rockwell International, Atomics International Division,
Rocky Flats Plant

R. F. Karlsson
M. A. Thompson
D. A. Wiederecht
R. E. Yoder

DISTRIBUTION (continued)

Sandia Laboratories, Albuquerque, New Mexico

L. W. Scully
W. D. Weart

Union Carbide Corporation, Nuclear Division,
Office of Waste Isolation, Oak Ridge

W. C. McClain
W. H. Pechin
C. D. Zerby

United Nuclear Industries

A. P. Larrick

Rockwell Hanford Operations

M. D. Alford
S. I. Anderson/E. W. Bartlett, Jr.
R. C. Arnett/R. G. Baca
N. M. Arnold
O. C. Asper
H. Babad
A. E. Barber
G. S. Barney/T. D. Cooper
D. C. Bartholomew
M. M. Beary/P. C. Ely
G. A. Beitel
C. R. Bergdahl
D. E. Braden
L. E. Bruns
J. S. Buckingham/T. A. Lane
L. A. Chick
D. J. Cockeram
R. A. Deju
P. A. Eddy/W. H. Riggsbee
R. E. Felt
M. R. Fox
R. D. Fox
P. J. Fritch
R. G. Geier
R. B. Gelman
R. E. Gephart
R. J. Gimera
D. R. Gustavson

DISTRIBUTION (continued)

Rockwell Hanford Operations

G. L. Hanson
D. G. Harlow
M. K. Harmon
J. K. Hartwell/D. L. Anderson
W. F. Heine
C. S. Homi
C. B. Honaker
H. H. Hopkins, Jr.

R. E. Isaacson
J. D. Kaser
D. E. Kelley
J. E. Kinzer
E. J. Kosiancic
M. J. Kupfer/D. M. Strachan
M. W. Legatski/D. J. Brown
R. Y. Lyon
C. W. Manry
P. F. Mercier
T. M. Nugent
R. G. Oliver
J. L. Peterson
K. R. Price/A. M. Schneider
S. M. Price
W. H. Price
I. E. Reep
R. C. Roal/W. Watson, Jr.
J. M. Roecker
W. L. Schick
W. W. Schulz/T. W. Lutton
H. P. Shaw
M. J. Smith/D. A. Dodd
J. S. Sprouse
C. P. Sutter
D. A. Turner
R. J. Thompson
H. A. Wallskog
R. A. Watrous/D. D. Chen
D. D. Wodrich
R. A. Yoder
Rockwell Document Services (3)
Central File