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ORNL/NFW-82/37

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**OAK
RIDGE
NATIONAL
LABORATORY**

**UNION
CARBIDE**

**ORNL Nuclear Waste Programs
Annual Progress Report for Period
Ending September 30, 1982**

**OPERATED BY
UNION CARBIDE CORPORATION
FOR THE UNITED STATES
DEPARTMENT OF ENERGY**

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ORNL/NFW-82/37

Contract No. W-7405-eng-26

**OAK RIDGE NATIONAL LABORATORY
NUCLEAR WASTE PROGRAMS**

Conventional Reactors
(Activity No. AG)

Remedial Actions
(Activity No. AH)

ORNL/NFW--82/37

Commercial Nuclear Waste
(Activity No. AP)

DE84 016535

Defense Waste
(Activity No. AR)

Spent Fuels
(Activity No. AS)

ORNL NUCLEAR WASTE PROGRAMS

ANNUAL PROGRESS REPORT FOR PERIOD ENDING SEPTEMBER 30, 1982

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May 1983

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Oak Ridge, Tennessee 37830
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HIGHLIGHTS

1. SPENT FUELS PROGRAM

A catalog was drafted and submitted to the Transportation Technology Center, Sandia National Laboratories for their review and approval covering such topics as federal, state, and local regulations, spent fuels characteristics, cask characteristics, transportation costs, and emergency response information.

2. AIRBORNE WASTE MANAGEMENT PROGRAM

2.2.1 Carbon-14 Immobilization

Carbon-14 immobilization development studies on the $\text{Ba}(\text{OH})_2 \cdot 8\text{H}_2\text{O}$ process have been completed. These studies were conducted at the micro-, bench-, and pilot-scale and will be the subject of two final reports, which are in preparation.

2.2.2 Filter Test Facility Quality Assurance

Three meetings of the Filter Test Facilities (FTF) Standards Writing Group were held, one at each facility. Final drafts of two standards were completed. Work began on preliminary drafts of two additional standards.

3. BYPRODUCTS UTILIZATION MANAGEMENT PROGRAM

Six major field tests of the radioluminescent lights were conducted this year in concert with development and fabrication. The field tests provided input from various Department of Defense services and the program expanded to include infrared lights and aircraft instrument lighting. Important improvements were made to the luminescent lights light output by proper selection of phosphors, application of phosphors to the tubes, tube geometry, and geometry of assembled fixtures.

A cesium-137 WESF capsule used in the Sandia sludge irradiator was examined for internal corrosion and capsule tensile strength.

Solubility testing of fully active cesium pollucite was completed.

4. SURPLUS FACILITIES MANAGEMENT PROGRAM

4.2.1 Defense-Surplus Facilities Surveillance and Maintenance

At the Metal Recovery Facility, defective roof hatches were replaced, radionuclide content of the canal was monitored, and a complete report on radiological characterization of the facility was issued. The stack was cleaned and cleared of debris at the ORNL Graphite Reactor in addition to regenerating demineralizer and routine filter replacements.

4.2.2 Curium Facility Decommissioning

Decontamination of the manipulator cells in the Curium Facility was completed by using a combination of techniques. Contamination levels in the cells are acceptable for the planned alternate use of the facility.

4.2.3 ILW Transfer Line Decommissioning

The abandoned transfer lines across the White Oak Creek floodplain and the high-pressure line from the old hydrofracture to the New Hydrofracture Facility were excavated and removed. The high-pressure pipe was salvaged and stored for reuse, and the abandoned pipeline was disposed of as waste.

5. HIGH-LEVEL WASTE MANAGEMENT PROGRAM

5.2.1 Concrete Grout Bulk Immobilization Study

ORNL demonstrated the feasibility of using grout bulk solidification for Hanford tank sludges and newly generated Purex wastes. Work is proceeding to establish the necessary grout formulations for specific wastes and to determine the properties of the resulting grouts.

6. INTEGRATED DATA BASE PROGRAM

A new issue of the annual report on spent fuel and radwaste inventories, projections, and characteristics was published, projecting to year 2020 and assuming reprocessing and defense HLW vitrification. Development of ORGENTRE-system codes continued for (a) nuclear materials (fissile and fertile) in the overall fuel cycle, (b) defense and I/I projections, and (c) waste treatment simulation. The ORIGEN2 code for isotope generation and depletion was maintained and updated for once-through fuel cycles. Bibliographic files were expanded and a bibliography published. The "Radioactive Waste Management Newsletter" was published (issues 15, 16/17, and 18).

7.1 DOE LOW-LEVEL WASTE MANAGEMENT PROGRAM

During FY 1982, the LLWMP Program Office at ORNL has drafted technology program plans, developed prioritized FY 1982, FY 1983, and FY 1984 budgets to support these plans, and provided technical guidance for the implementation of these plans at participating DOE sites.

8. WASTE MANAGEMENT OPERATIONS PROGRAM

8.2.1 Operational Planning and Development

An ad hoc committee for development of a long-range plan for radioactive waste management completed a draft report that describes the various waste systems, discusses problems and shortcomings, and recommends improvements in facilities and practices. The committee's report will be issued in two parts during FY 1983.

The New Hydrofracture Facility was completed. A dedication ceremony was held, and the facility became operational in June. Construction of the new building for the Waste Operations Control Center was completed, and a contract for procurement of the data acquisition system was negotiated. Demolition of the old precipitator at the 3039 Stack Off-Gas System and modifications to the scrubber were completed, and construction of the temporary by-pass loops is in progress. Major construction at the new flow monitoring station No. 3 on White Oak Creek was completed, and work on the other two stations is progressing on schedule. Progress on General Plant Projects included completion of four clusters of piezometer wells in SWSA 5 and a monitoring weir for the Northwest Tributary of White Oak Creek. Installation of the electro-polishing equipment at FPDL was completed and a core storage building was constructed. Procurement and installation of a fourth column for the Process Waste Treatment Plant has been delayed because the column did not pass inspection and testing.

Liquid and Gaseous Wastes System Operations

After preoperational testing, including a cold injection in March, the New Hydrofracture Facility (NHF) began hot operation in June with injection of about 606 m^3 (160,000 gal) of concentrated intermediate-level waste. Also, two injections of resuspended gunite tank sludge were made: in August about 696 m^3 (184,000 gal) and in September about 454 m^3 (120,000 gal).

8.2.3 Solid Waste Storage Management

About 54 m^3 ($1,910 \text{ ft}^3$) of TRU waste were stored retrievably. About $1,886 \text{ m}^3$ ($66,563 \text{ ft}^3$) of beta-gamma waste were received: $1,404 \text{ m}^3$ ($49,598 \text{ ft}^3$), including 34 m^3 which was 278 m^3 before compaction, were buried in trenches and auger holes and 238 m^3 ($8,403 \text{ ft}^3$) in landfill for low-hazard contaminated waste.

8.2.4 Guniting Tank Sludge Removal Project

Construction of the guniting tank sludge removal project was completed. Necessary documents were prepared, reviewed and approved. Preoperational testing was completed and sluicing began on June 12. As noted above, two injections of resuspended sludge were made at the NHF: in August, about 696 m^3 ($184,000 \text{ gal}$) and in September about 454 m^3 ($120,000 \text{ gal}$).

8.2.5 Site Specific Low-Level Waste Studies

A detailed site characterization of the geochemical, hydrological, and geological factors governing radionuclide migration is presently being undertaken at ILW Trench 7. This site was chosen because of the occurrence of ^{60}Co and ^{233}U in a surface seep which appears in a valley 50 m east of the trench. Groundwater ^{60}Co and ^{233}U concentrations undergo a seasonal variation at this site with the highest activities occurring in the springtime. The seasonal rise in nuclide activity is correlated with a rise in the groundwater level and a concurrent increase in groundwater pH. This suggests that during winter and spring, the water table may rise to saturate the alkaline fill in the trench or contaminated soils near the trench and thus promote nuclide migration. A new series of wells and soil borings is presently being drilled at the north end of the trench to evaluate the above mechanism and pathway for nuclide mobilization and

transport and the effectiveness of the groundwater interceptor drain as a remedial action plan to retard nuclide transport along this pathway. Since SWSA 4 is the most important contributor of ^{90}Sr to White Oak Creek, it has been the focus of detailed studies of mechanisms for the hydrologic transport of ^{90}Sr from disposal trenches. These studies have shown that surface runoff is of major importance at SWSA 4 and have resulted in plans for a surface water diversion to capture uncontaminated runoff from the upslope portions of the SWSA 4 watershed and divert it. In the meantime, soil and water samples have been taken to characterize the nature and extent of radionuclide contamination that is potentially susceptible to transport by groundwater runoff. The nature of the three-dimensional distribution pattern strongly suggests that the spread of contamination results from the "bathtub" effect within several individual trenches in low-lying areas. These trenches, which were constructed perpendicular to contour lines, collect water and overflow to the ground surface at the downslope end rather than releasing contaminated water to the surrounding soils for subsequent groundwater transport. It is expected that this shortened pathway will be reduced to a minor role when the surface water diversion construction is completed.

Dye tracer experiments have confirmed that SWSA 3 is the source of radiostrontium to seeps on Raccoon Creek and the Northwest Tributary (NWT) to White Oak Creek. The relative intensity of dye reaching the NWT and the Raccoon Creek seep indicates that a spring on the NWT is the major recipient of radionuclides migrating from SWSA 3. This spring should be an excellent monitoring point for assessing the effectiveness of any remedial actions implemented at SWSA 3. This work is of particular significance because SWSA 3 overlies a limestone formation that is known to contain solution openings.

8.2.6 Hydrofracture Mix and Grout Development Studies

Studies showed no difference between the performance of the suspending agent Attagel-350 and the Attapulgate-150 drilling clay in hydrofracture grouts.

Dry solids blending tests showed the necessity of homogenous blending to assure acceptable grout rheologies.

Microscopic examination of an 18 year old grout core showed no radionuclide migration into the shale.

8.2.7 Waste Volume Reduction Operations

Two methods of dissolution of the aluminum metal-europium oxide HFIR control plates were tested successfully.

An electropolisher and a vibratory finisher were installed at the Fission Product Development Laboratory to decontaminate metallic tools and small equipment for reuse at ORNL facilities or for salvage operations. The reduction in the volume of this type of material presently sent to the burial ground will aid in the conservation of the limited burial ground space available at ORNL.

Laboratory studies indicate viability of using anaerobic digestion technology for volume reduction of low-level solid radioactive waste generated at ORNL. The volume reduction factor for this process is expected to be approximately 20:1.

8.2.8 Central Waste Disposal Facility Design and Development

A single large area on West Chestnut Ridge, near ORNL, was selected as the reference site for a central waste facility for LLSRW disposal. Preliminary field studies (geological, hydrological, and seismic refraction) were completed on six topographically attractive tracts. As a result of these studies, a decision was made to continue examining

this site with more detailed field studies. Because the reference site could not yet be accepted with a high degree of confidence, however, a decision was made to select an additional, alternative site on which to begin preliminary field studies.

For ORNL, the central waste facility will complement SWSA 7, the solid waste storage area being developed for low-volume, higher-activity solid waste.

8.2.9 Metal Waste Management at ORO Sites

The Conceptual Design Requirements for a Metal Smelting Facility report was completed during FY 1982. Various disposal options for the ORGDP scrap metal were also investigated and a complete report was issued. Work was completed and a report was issued which detailed the potential need within the government for the steel produced from smelting of the contaminated scrap metal. An additional study has been completed, and a report issued, that investigated processing the contaminated scrap metal using commercial metal processing and smelting firms for both on-site and off-site operations.

8.2.10 Traffic Managers Support

A Transportation of Hazardous Materials: Directory of Principal Contacts was developed designating the agencies/personnel responsible for transporting hazardous/radioactive materials and emergency response contacts for each state and locale.

The data base for the HIGHWAY routing model has been upgraded to include principal DOE sites, nuclear reactors, and major commercial airports. A special program has been developed for identifying the airports in the vicinity of any location included in the HIGHWAY data base.

9. TRANSPORTATION PROGRAM

9.2.1 Defense Waste Transportation Studies

The Legislative and Regulatory Information System was divided into a current and a historic data base, both of which were converted to the new data management system, INQUIRE. The Seventh International Symposium on Packaging and Transportation of Radioactive Materials has been scheduled in New Orleans, Louisiana, in May 1983. Instructions for the preparation of papers have been completed. A study to compare the use of various criticality codes by the International Committee was completed and results indicated that codes employing Monte Carlo and other transport methods give satisfactory results.

9.2.2 TRUPACT Transport System Tests

Newly designed TRU waste packages, intended to be carried in the TRUPACT container (an overpack currently being designed by the Transportation Technology Center at Sandia National Laboratory) were impact tested under conditions specified in DOE and NRC regulations.

10. TRU WASTE MANAGEMENT PROGRAM

10.2.1 FUETAP Concrete Immobilization

The first FUETAP process proof-of-principal tests at an engineering scale (1/4-ton) was successfully completed.

ORNL formulated and tested a cement-based waste host from the Rocky Flats Plant Building 374 sludges.

10.2.2. Instrumentation Prototype

To date 114 drums have been assayed and gamma-scanned by the NIAS and segmented drum scanner, respectively. We have tentatively identified 46 drums as being below the 100 n Ci/g limit for TRU waste and 55 drums as being over the limit. Maximum throughput of drums in the active and total coincidence neutrons assay modes (not including neutron multiplicity measurements and gamma-scanning) was found to be six to seven drums per hour (40-50 drums per 8-hr shift) with sufficient personnel and equipment.

Destructive assaying of the drums will entail the transfer of a drum's contents into a glove box for examination, segregation of waste type, shredding of cellulosics, and plastics, and sampling for gamma-ray spectroscopic analysis. It is planned to assay approximately 100 drums taking approximately 100 samples from each drum.

10.2.3 TRU Waste Certification Program

A certification document for handling newly generated contact handled transuranic waste at ORNL was completed and internally reviewed. All of the certification document, except the detailed site handling procedures, has been informally submitted to TWSO for comment.

10.2.4 Cementitious Waste Form Development for PREPP

Preliminary formulas for high- and low-combustible ashes from the PREPP at INEL were formulated and proof-of-principle tests of the cementation step were supported on site. Recommendations regarding mixing equipment and accelerated testing for the cemented wastes were made.

11. HIGH-LEVEL WASTE MANAGEMENT PROGRAM

11.2.1 ORNL Fixation of Wastes in Concretes

A draft of the close-out evaluation report on the use of autoclaved concretes for high-level wastes was prepared.

12. INTEGRATED DATA BASE PROGRAM

12.2.1 Integrated Data Base Program

A new issue of the annual report on spent fuel and radwaste inventories, projections, and characteristics was published, projecting to year 2020 and assuming reprocessing and defense HLW vitrification. Development of ORGENTRE-system codes continued for (a) nuclear materials (fissile and fertile) in the overall fuel cycle, (b) defense and I/I projections, and (c) waste treatment simulation. The ORIGEN2 code for isotope generation and depletion was maintained and updated for once-through fuel cycles. Bibliographic files were expanded and a bibliography published. The "Radioactive Waste Management Newsletter" was published (issues 15, 16/17, and 18).

13. LOW-LEVEL WASTE MANAGEMENT PROGRAM

13.2.1 Low-Level Waste Information Center

The low-level waste information center received 200 new documents, wrote 150 abstracts, and responded to 300 information requests on technical information relevant to low-level radioactive waste technology during the funding period October 1, 1981 to February 1, 1982.

13.2.2 State Screening Project

A report has been drafted for the State of Tennessee describing a process of identifying areas that have the greatest probability of containing acceptable sites for shallow land burial of low-level radioactive waste.

14. TRANSPORTATION PROGRAM

14.2.1 Commercial Waste Transportation Studies

Approximately 400 new bills pertaining to the transportation of radioactive materials were added to the Legislative Data Base.

The Test and Evaluation Facility Logistics Study was completed for the Transportation Technology Center, Sandia National Laboratories. The NUS waste management model RAWSYM was successfully transferred to and tested on the ORNL computer system.

15. TRU WASTE MANAGEMENT PROGRAM

15.2.2 TRU Commercial Fuel Cycle Studies

TRU wastes to be expected from the commercial reprocessing of LWR fuel were identified and quantitatively listed, based on an AGNS-type flowsheet, and for a 1500 MTU/y plant. Isotopics were based on 16 year old fuel at an average burnup of 25,200 MWd/MTM. Qualitative differences for the reprocessing of LMFBR fuel were identified.

15.2.3 TRU Isotope Decay Chains

The detailed alpha decay behavior of 38 isotopes, including daughters, were computed and summed using ORIGEN2. The results were then summarized and presented graphically for the time span 0.1 to 10^6 y.

16. WASTE ISOLATION PROGRAM

16.2.1 Regional Environmental Characterization Studies - Nuclear Waste

This task supports the ONWI overall goal of providing a significant degree of assurance that high-level nuclear wastes can be isolated from the biosphere in a safe and environmentally acceptable manner.

Information contained in the regional environmental characterization report was used to screen geologic formations which have potential as high-level nuclear waste repositories to select formations suitable for further detailed analyses in the area phase of this study.

16.2.2 Institutional Arrangements for Nuclear Waste Siting

There is considerable uncertainty that remains in the repository design and siting process indicating a need for continued research to lessen uncertainty, and flexibility in job training program design as uncertainty is reduced. There are likely to be multiple major factors that need to be involved in job training program design. There is a need for development of mechanisms for resolving potential conflicts among these major actors. Program design and implementation decisions need to be documented carefully to protect each program participant's interest. A clear assignment of responsibilities for program design and implementation must be made.

16.2.3 Evaluation of Avery Island Coupons

Tests to obtain the necessary corrosion data so that various multi-barrier waste package emplacement concepts in domal salt could be evaluated were performed and showed that only carbon steel experienced appreciable attack.

16.2.4 Effects of Water in Salt Repositories

Through experimental and modelling studies, we are attempting to predict the consequences of the presence of water in a salt repository that might compromise the integrity of the repository. At this time, testing is not complete.

16.2.5 Salt Model Pillar Studies

Careful examination of the creep records of the long-term tests revealed the existence of episodes of transient creep occurring between periods of steady behavior. The potential importance of these transient creep episodes in repository design and in the verification of computer codes to be used in design justified a more in-depth investigation.

16.2.6 Expected Repository Environments for Commercial Wastes, Savannah River, and Other Wastes

A series of reports has been issued describing the environments providing information in designing experiments and repositories and in licensing procedures based on calculated expected repository environments for the disposal of both commercial and defense high-level wastes and spent fuel in salt formations.

16.2.7 Technical Topical Report on Brine Migration

A report was issued describing the technical findings on brine migration in a salt repository as input to a licensing topical report.

16.2.8 System Model to Assess Flooding in a Waste Repository

In order to develop the methodology and appropriate computer codes for analyzing the effects of freshwater intrusion into a salt repository for nuclear waste that also has applicability for other geologic media, salt dissolution mechanisms were examined and the sequence of solutioning

was divided into two phases. Models for the two phases were identified. The literature on leaching of radionuclides was examined and appropriate models were identified. Evaluation of the experimental data on the permeability of crushed salt as a function of time while being stressed was completed. Considerable progress was made on developing a computer code for simulating the flooding process. This code (REFS) will be based on the existing code HYTAQ.

16.2.9 Sensitivity and Uncertainty Analysis Methods Development

Development of the GRESS computer language to automatically calculate the sensitivities of waste repository model performance with respect to model data results will be central to the development of a comprehensive methodology for sensitivity and uncertainty analysis for waste repository design licensing. This task began in June 1981 with the development of a demonstration version of the GRESS automated language approach to adjoint sensitivity analysis. The program was halted September 30, 1981, with the completion of GRESS Versions A. After an evaluation, plans were made for full-scale development and implementation of this approach for repository design licensing.

17. SURPLUS FACILITIES MANAGEMENT PROGRAM

17.2.1 Commercial Surplus Facilities Surveillance

The annual reheat of the fuel in the Molten Salt Reactor Experiment (MSRE) was completed in addition to routine inspection and changing of filters. A radiological survey of the Homogeneous Reactor Experiment (HRE-2) was initiated and completed except for soil core samples. A leaking shield plug previously used for a loop experiment at the Oak Ridge Reactor (ORR) was replaced; the defective shield was disposed of as waste.

18. MILL TAILINGS PROGRAM

18.2.1 Liner Evaluations for Uranium Mill Tailings

A draft report was completed on the assessment of in situ grouting as an alternate remedial action for mill tailings piles. The topical report will be issued as soon as internal review is complete.

18.2.2 UMTRAP Information Support

Information support was supplied to an internal Symposium and by responding to special requests. A third volume of the bibliography entitled "Nuclear Facility Decommissioning and Site Remedial Actions" was prepared and issued.

19. FORMERLY UTILIZED SITES REMEDIAL ACTIONS PROGRAM

The concept of in situ stabilization has been proposed for both the Middlesex Sampling Plant Site and other facilities under FUSRAP. Expressed in the most simplified manner, this concept envisions utilizing the natural materials (soil, other sediments, bedrock where applicable) at a site to contain the radionuclides present in contaminated material. This can be possibly accomplished by either admixing these natural earth materials with the wastes or burying the wastes within an "envelope" provided by the natural materials. In some cases, both admixing and burial can be practiced to assure containment. In a related approach, various substances or additives, either natural such as bentonite or artificial such as ion-exchange resins, can be added to further the containment capabilities of the natural materials available at a given site.

20. THREE MILE ISLAND PROGRAM

20.2.1 Technical Advisory Group Activities

The Technical Advisory Group activities focused on the performance of the Submerged Demineralizer System for containment building and reactor coolant system water cleanup. A number of other problems involving aqueous stream decontamination were also addressed.

20.2.2 Technical Assistance and Advisory Group

Participation on TAAG involved reviews of many significant TMI-2 recovery operations including the axial power shaping rod motion tests and the "quick look" video camera insertion.

20.2.3 Technical Support Activities

During this fiscal year, the technical support activities included an examination of expected source terms during refueling, studies of criticality during recovery, evaluation of TMI-steam generator tube corrosion, studies of the elution of a cesium-loaded zeolite column that is currently a significant radioactivity source in the containment building, development of a method for chloride removal from RCS water, examination of possible causes for failure of the self-powered neutron detectors at TMI-2, studies of the rate of uptake of cesium and strontium by specific zeolites, and various chemical analyses.

20.2.4 Development of Inorganic Sorbents

Eighteen candidate inorganic sorbents were screened for possible use in the decontamination of aqueous streams likely to be encountered in light water reactor systems. This activity was terminated at the conclusion of the screening tests, however, in order to focus attention on more urgent problems associated with TMI-2 recovery.

SPENT FUELS PROGRAM

1. SPENT FUELS PROGRAM

L. B. Shappert

1.1 INTRODUCTION

The goal of this task is to support the Domestic Spent Fuel Storage Program through studies involving the transport of spent fuel. The major objective is to develop a transportation catalog for spent fuel.

1.2 PROGRESS

Technical progress for the task of this program for the fiscal year just ended follows.

1.2.1 Spent Fuel Storage (L. B. Shappert)

A catalog was developed to provide authoritative, timely, and accessible transportation information for persons involved in the transport of irradiated reactor fuel. The catalog, drafted and submitted to the Transportation Technology Center, Sandia National Laboratories, for their review and approval, covers such topics as federal, state, and local regulations, spent fuel characteristics, cask characteristics, transportation costs, and emergency response information.

1.3 PUBLICATIONS

None.

1.4 REFERENCES

None.

2. AIRBORNE WASTE MANAGEMENT PROGRAM

W. W. Pitt

2.1 INTRODUCTION

The airborne release of ^{14}C from various nuclear facilities has been identified as a potential bio-hazard due to the long half-life of ^{14}C (5730 years) and the ease with which it may be assimilated into the biosphere. The objective of the ^{14}C Immobilization task is to develop technology for the removal and immobilization of this radionuclide. Prior studies have indicated that ^{14}C will likely exist in the oxidized form as CO_2 and will contribute slightly to the bulk CO_2 concentration of the gas stream, which is airlike in nature ($\sim 350 \text{ ppm}_v \text{ CO}_2$). The technology under development utilizes the $\text{CO}_2\text{-Ba(OH)}_2\cdot 8\text{H}_2\text{O}$ gas-solid reaction with the mode of gas-solid contacting being a fixed bed. The product, BaCO_3 , possesses excellent thermal and chemical stability, prerequisites for the long-term disposal of nuclear wastes. For optimal process operation, studies have indicated that an operating window of adequate size does exist. When operating within the window, high CO_2 removal efficiency (effluent concentrations $< 100 \text{ ppb}_v$), high reactant utilization ($> 99\%$), and an acceptable pressure drop across the bed (3 kPa/m at a superficial velocity of 13 cm/s) are possible. The experimental program has addressed three areas of experimental investigation: (1) microscale studies on 150-mg samples to provide information concerning surface properties, kinetics, and equilibrium vapor pressures; (2) macroscale studies on large fixed beds (4.2 kg of reactant) to determine the effects of humidity, temperature, and gas flow rate upon bed-pressure drop and CO_2 breakthrough; and (3) design,

construction, and initial operation of a pilot unit capable of continuously processing a $34 \text{ m}^3/\text{h}$ ($20\text{-ft}^3/\text{min}$) air-based gas stream.

The objective of the Filter Test Facilities Quality Assurance task is to develop a set of standardized procedures for the operation and guidance of the three DOE Filter Test Facilities at Richland, Washington; Rocky Flats, Colorado; and Oak Ridge, Tennessee; to analyze operating data from the three facilities to determine trends with respect to operating loads and quality of HEPA filters being supplied to DOE contractors; and to provide general technical assistance to the Filter Test Facilities and to the DOE Airborne Waste Program.

2.2 PROGRESS

Technical progress for the Carbon-14 Immobilization task and Filter Test Facility Quality Assurance task for FY 1982 follows.

2.2.1 Carbon-14 Immobilization (G. L. Haag, J. W. Nehls, Jr. and W. G. Chapman)

Extensive studies on $\text{Ba}(\text{OH})_2$ hydrates, their reaction with CO_2 , and the operation of fixed-bed processes for CO_2 removal from air-based gas streams ($330 \text{ ppm}_v \text{ CO}_2$) were continued and the results analyzed. Macro-scale studies have indicated that (1) the published vapor pressure for $\text{Ba}(\text{OH})_2 \cdot 8\text{H}_2\text{O}$ is valid, (2) the rate of dehydration of $\text{Ba}(\text{OH})_2 \cdot 8\text{H}_2\text{O}$ to $\text{Ba}(\text{OH})_2 \cdot 3\text{H}_2\text{O}$ and the subsequent rehydration at relative humidities $<55\%$ is proportional to the amount of free water on the surface (i.e. a function of relative humidity), and (3) the reactivity of $\text{Ba}(\text{OH})_2 \cdot 8\text{H}_2\text{O}$ for CO_2 is 3 orders of magnitude greater than that of either $\text{Ba}(\text{OH})_2 \cdot 3\text{H}_2\text{O}$ or $\text{Ba}(\text{OH})_2 \cdot \text{H}_2\text{O}$. Macroscale studies under near-isothermal conditions on 10.2-cm-ID fixed beds of commercial $\text{Ba}(\text{OH})_2 \cdot 8\text{H}_2\text{O}$ flakes (reference

superficial velocity of 13 cm/s) indicated that the pressure drop across the bed increased dramatically as 55% relative humidity in the influent gas was approached. The dependency upon relative humidity rather than water vapor pressure was confirmed by the consistency of data obtained at two temperatures, 22°C and 32°C (a difference in the water vapor saturation pressure of 15.8 mm Hg). Similar studies conducted with the ^{14}C Immobilization pilot unit under near-adiabatic conditions indicated a similar increase in pressure drop when the influent relative humidity approached 55%. Because of the adiabatic nature of the reaction, the temperature of the gas stream decreased by $\sim 4^\circ\text{C}$ upon passing through the bed. It is speculated that the increase in pressure drop results from the capillary condensation of water at V-shaped contact point or pores within the solid. The presence of the condensed water incurs two phenomena. First, the substoichiometric flakes hydrate and recrystallize upon forming $\text{Ba}(\text{OH})_2 \cdot 8\text{H}_2\text{O}$. During the step, the flakes tend to curl and although the surface area is increased, the flakes are more fragile. Second, studies have indicated that the surface area of the product decreases as the relative humidity increases. This would indicate a subsequent increase in crystallite size with decreasing relative humidity. We speculate that the larger crystallites are more likely to cleave from one another and that this results in flake degradation and increased pressure drop. This information will be addressed in greater detail in a final report.

Experimental studies indicated that the transfer of the reactant gas through the gas film is the major resistance to mass transfer. A model, assuming gas film control, was developed and exact numerical solutions were obtained. An excellent correlation between the model-predicted breakthrough curves and the experimental breakthrough curve was obtained

when the area available for mass transfer was modeled as a linear function of conversion [i.e. $A = A_0(1 - X)$]. The magnitude of the mass transfer coefficient was characteristic of literature values. There were indications that the magnitude of the initial surface area available for reaction, A_0 , may be weak function of velocity due to a realignment of the flakes. This realignment results from fluid shear forces and an accompanying reduction in the number of planar contact points between neighboring flakes; thus, increasing the area available for mass transfer.

Based upon the experimental data obtained during this study and its subsequent analyses, a window or regime of optimal process operation under near-isothermal and near-adiabatic conditions was determined to exist for the fixed-bed process. The window is bounded on the lower side by the dissociation vapor pressure of $\text{Ba}(\text{OH})_2 \cdot 8\text{H}_2\text{O}$ and on the upper side by the onset of appreciable capillary condensation and subsequent pressure drop problems (~55% relative humidity). An operating envelope is presented in Fig. 2.1 for the treatment of a 330-ppmv- CO_2 gas stream. The relative humidity of the influent gas must fall within the envelope for optimal gas throughput. The operating envelope also demonstrates why operational problems at 22 and 32°C were not severe and why considerable difficulty was encountered when trying to operate the process at 42°C.

In the development of this fixed-bed technology, a pilot unit capable of processing 34 m^3/h (20 ft^3/min) was designed, constructed, and operated. Specific goals of this aspect of process development were to provide (1) the basis for the design of a ^{14}C immobilization module for future testing under hot conditions; (2) data at operating conditions not achievable with present bench-scale equipment, in particular, operation at

Fig. 2.1. Operating window for contacting a 330-ppmv CO_2 gas stream with fixed beds of commercial $\text{Ba}(\text{OH})_2 \cdot 8\text{H}_2\text{O}$ flakes.

near-adiabatic conditions; (3) necessary scale-up data; and (4) operating data on key hardware items and instrumentation.

A flow-instrumentation diagram of the ^{14}C Immobilization pilot unit is shown in Fig. 2.2. A photograph of the system is presented in Fig. 2.3. The designed gas throughput at a superficial velocity of 13 cm/s in the reactor is $34 \text{ m}^3/\text{h}$ ($20 \text{ ft}^3/\text{min}$). The system consists of two reactors which contain canisters loaded with 32 kg (70 lb) of commercial $\text{Ba}(\text{OH})_2 \cdot 8\text{H}_2\text{O}$ reactant. Due to the size of the canisters and the relatively long loading times prior to breakthrough, continuous operation with only two reactors is possible. Approximately 2100 hours of operating time (8 cell loadings) were accumulated prior to the termination of the experimental work. Detailed results of these studies will be presented in a final report.

2.2.2 Filter Test Facility Quality Assurance (C.A. Burchsted)

2.2.2.1 Standards

Three meetings of the Filter Test Facilities (FTF) Standards Writing Group were held during the year, one at each station. Although earlier plans called for a family of eight standards to provide the required guidance for FTF operations, this was reduced to four by combining several related topics into one standard, NE F 3-43. Final drafts of two standards, NE F 3-42, Operating Policy for DOE Filter Test Facilities, and NE F 3-43, DOE Filter Test Facility Operating Procedures, were completed and submitted to the Technical Review Committee for preliminary review prior to balloting. Preliminary drafts of two additional standards, one a guide for preparing quality assurance programs for the FTFs and the second

Fig. 2.2. Flow-instrumentation diagram of the ^{14}C immobilization pilot unit.

Fig. 2.3. The ^{14}C immobilization pilot unit.

a guide for procurement of HEPA filters by DOE contractors, were started and will be presented to the Writing Group early in FY-1983.

2.2.2.2 Round Robin

Procedures for round robin tests to validate FTF operations were completed and incorporated in NE standard F 3-43. The initial round robin was started at the end of the fiscal year and will be complete during the first quarter of FY-1983. Round robins will be conducted at least annually in the future.

2.2.2.3 Operating Data Analysis

A review of FTF operating data for the years 1960 (initiation of FTF program) through 1980 was completed. Findings were reported to DOE and as a paper at the 17th DOE Nuclear Air Cleaning Conference. With a change of emphasis and the addition of data for 1981, the findings were also reported at the World Filtration Congress III.

2.3 PUBLICATIONS

C. A. Burchsted, "Experience in the Quality Assurance Testing of HEPA Filters," pp 15-22 in the Proceedings of World Filtration Congress III, Sept. 13-17, 1982.

G. L. Haag, Application of the $\text{CO}_2\text{-Ba(OH)}_2\cdot 8\text{H}_2\text{O}$ Gas-Solid Reaction for the Treatment of Dilute CO_2 -Bearing Gas Streams, Ph.D. Dissertation, University of Tennessee, Knoxville; and ORNL-5887 (in preparation).

G. L. Haag, J. W. Nehls, Jr., and G. C. Young, "Carbon-14 Immobilization via the $\text{Ba(OH)}_2\cdot 8\text{H}_2\text{O}$ Process," Proceedings of the 17th DOE Nuclear Air Cleaning Conference, Denver, CO., Aug. 2-5, 1982.

C. A. Burchsted, "A Review of DOE Filter Test Facility Operations, 1970 Through 1980," Proceedings of the 17th DOE Nuclear Air Cleaning Conference, Denver, CO., Aug. 2-5, 1982.

2.4 REFERENCES

None.

3. BYPRODUCTS UTILIZATION MANAGEMENT PROGRAM

S. M. Tiegs

3.1 INTRODUCTION

Field testing of radioluminescent lights was conducted in concert with development and fabrication. The field tests provided input from various Department of Defense services and the program was expanded to include infrared lights and aircraft instrument lighting. As a consequence, participation by the Army has become an important focal point for extended utilization, especially by the rapid deployment forces. Important improvements were made to increase light output by proper selection of phosphors, application of phosphors to the tubes and tube geometry.

A cesium-137 WESF capsule used in the Sandia sludge irradiator was examined for internal corrosion and capsule tensile strength. Solubility testing of fully-active cesium pollucite was completed.

3.2 PROGRESS

Technical progress for each of the tasks of this program for fiscal year 1982 follows.

3.2.1 Terrestrial Radioisotope Application Development (F. N. Case, K. W. Haff, J. A. Tompkins, and F. J. Schultz)

3.2.1.1 Radioluminescent Light Source Development

3.2.1.1.1 Phosphor Development. A survey was made to obtain candidate samples of phosphors for testing with tritium to maximize light output. Eighteen phosphor samples were obtained and tested for light output when exposed to tritium. From this group, one yellow-green and one infrared phosphor were selected for use in fabrication of light tubes. The yellow-green phosphor increased light output approximately 20% over the best phosphor used in earlier work. Phosphor particle size has been identified as an important parameter in light output; however, work has not progressed to the point of selection of an optimum size for a given phosphor.

Phosphor coating thickness control and proper composition of binders for the application of phosphors to glass tubes are required to obtain

uniform coatings of phosphors on tubes. In our earlier work with binders we used a silicone binder that provided a highly stable adhesive for phosphor coatings. However, this preparation was a proprietary product and has been discontinued. Work was then conducted using the conventional phosphoric acid binders, and satisfactory methods for coating tubes were developed. Decomposition of the phosphor when it is exposed to phosphoric acid occurs to a limited extent at the phosphor/glass interface. A formulation consisting of 0.5% phosphoric acid in a 95% acetone, 5% hexyl alcohol mixture was found to bind the phosphor to well-cleaned and hydrofluoric acid-etched glass.

Ten phosphor manufacturers and suppliers in the U.S., Europe, and Japan were contacted in a search for more efficient phosphors for use with tritium and krypton-85. The search resulted in a phosphor with a 20% greater light output.

3.2.1.1.2 Fabrication and Light Fixture Development. Runway distance markers were fabricated and supplied to Tyndall Air Force Base for testing and evaluation. These 4 x 4 ft square markers were assembled and shipped to Tyndall for installation. Each marker contained flat glass light tubes 1/4 in. thick x 3/4 in. wide that varied in length from 4 to 22 in. Thirty individual units were fabricated that comprised both runway distance markers and taxiway information signs. (Figure 3.2.1)

ORNL-PHOTO 5717-80

Figure 3.2.1 Typical Tritium Taxiway Sign

Approximately 150 runway edge lights comprised of 120 visible and 30 infrared lights were fabricated and tested. These units consisted of three individual pyrex light tubes containing 32 Ci of tritium each and packaged inside a 1-1/2 in. diameter polycarbonate tube sealed unit. Each unit had 10 in. of light projection over its 15 in. of total length. In use, two units are deployed on a "tee" shaped holder (Figure 3.2.2) placed in the ground at the edge of an airfield runway. This arrangement makes it easy to deploy the units and provides for 6 in. of spacing between individual lights on the holder. Thus, the apparent light size is increased, making the acquisition distance greater than is possible with individual lights.

ORNL-PHOTO 3607-82

Figure 3.2.2 Tritium Lights Assembled on "Tee" Holder

A small fixture containing three 6 in. long light tubes assembled inside of a 1-1/2 in. diameter polycarbonate sealed tube and equipped with a handle and a stud bolt for attaching several fixtures end to end was designed and tested. These fixtures were developed to meet a need by rapid deployment forces for use in marking drop zones and for identification of combat-important ground locations. Both visible and infrared phosphors were used in these fixtures. Field tests are scheduled for FY 1983.

3.2.1.1.3 Ruggedizing. Military use of radioluminescent lights requires ruggedized systems to permit air dropping, rough handling, and exposure to environmental extremes. Clear polycarbonate provides more resistance to breakage than any of the transparent construction plastics available and is used to enclose the pyrex glass tubes containing the phosphor and radio-nuclide. The tubes are shock mounted on silicone rubber foam spacers and

resist breakage under normal handling and shipping conditions. High-speed photography was used to observe deflections in the pyrex tubes that occur during mechanical shock conditions. The photographic observations are presently being evaluated. The objective of this effort is to develop methods for testing and diagnostic examination that will permit the fabrication of fixtures that will meet military combat requirements.

3.2.1.1.4 Physical Testing. Testing of light fixtures and shipping cartons to meet ANSI and DOT standards was carried out. Two lighting systems were tested. Runway distance markers were tested to meet the ANSI standard for self-luminous signs, and an NRC license for use of the markers at Tyndall Air Force Base was obtained. The hand-held lights, consisting of three tubes in a polycarbonate fixture, were tested according to ANSI standard tests and passed. In addition, a percussion test was carried out to determine the impulse required to distort the polycarbonate cover tube to the point of breakage of the internal pyrex light tubes. Shipping containers were procured and Type A package certification testing was conducted.

3.2.1.1.5 Light Measurement. Light output measurement is required for evaluation of phosphors and to optimize geometry for fabrication of light fixtures. Equipment was installed in a totally dark, non-reflective walled area. This equipment consists of an EG&G, Model 550 Photometer. One hundred fifty candella measurements were made for krypton-85 lights fabricated and tested for runway edge lights and for taxiway lights. A typical isocandella profile is shown in Figure 3.2.3. Infrared light output was obtained using this photometer also.

3.2.1.1.6 Field Testing. The major effort for the year involved fabrication of fixtures and field testing for military applications. This was a highly productive effort from the standpoint of introducing the military to the potential of radioluminescent lights as a reliable and long-lived light source for low-level lighting. Suggestions from user personnel provided the direction needed to make the program functional and to expand participation of various military services.

Ft. Rucker Alabama Test. A demonstration of visible lights, with both krypton-85 and tritium lights, was given during a General staff conference

Figure 3.2.3 Isocandella Curves for a Krypton-85 Light Source with Bi-directional Light Pipe

at Ft. Rucker, Alabama, in March 1982. This demonstration provided several General officers with a first hand observation of the lights and their potential applications. It also permitted reviewing with night vision equipment by the conference attendees. A full test in May 1982, also at Ft. Rucker, was conducted to demonstrate helicopter landing field marking. This test used single three-tube tritium light wands set out in a line extending from each end of two parallel runways to provide alignment for aircraft maneuvers. Lights were deployed for five nights and used in training exercises. Both visible and infrared lights were used in this test. Pilot evaluation was favorable and the lights were judged to be adequate for lane marking in remote field applications.

Pope Air Force Base, North Carolina. A field test was conducted at Pope Air Force Base, North Carolina to compare visible and infrared radioluminescent lights with battery operated units. The battery operated infrared units were judged to be about two times brighter than the radioluminescent lights. The radioluminescent lights, however, appeared to be adequate for most requirements and have the added advantage of reliability not possible with battery operated units. The radioluminescent lights have many potential military applications. At the field test it was suggested that paratrooper wrist altimeters be fitted with tritium lights. Two units were obtained and illuminated with tritium light sources for evaluation.

Naval Ship Yard - Connecticut. A test was conducted to determine the visibility of a standard three-tube light under conditions of murky waters as might be encountered during a ditching of a helicopter from the deck of a ship. The objective was to provide an exit light that is unaffected by electrical power failure to orient the pilot of the aircraft, in or under water. Further testing was judged to be necessary with increased intensity of light or one of different spectral quality.

Fort Huachuca, Arizona. A planning session and a field test were conducted at Fort Huachuca, Arizona to test use of lights under desert conditions and in a low-ambient light environment. The test was set up at a simulated base airfield of the type used by advance combat rapid deployment forces. One hundred twenty lights were fabricated and deployed as runway edge lighting and threshold lighting. Helicopter and Mohawk aircraft were used in this test. The lighting system was reviewed by Army rapid deployment officers and enlisted personnel and was judged to be acceptable as a

landing aid when used with radar guidance for field location. Recommendation of a second bare base test to be made in Hawaii was put in form of an Army request by the Hawaiian base commander who was visiting Ft. Huachuca. A glide slope indicator was set up using both visible and infrared lights and was used successfully by helicopters.

Marine Corps Auxiliary Landing Field, Bogue, North Carolina. A demonstration of runway edge lighting and classified infrared tests were conducted at the MCALF, Bogue, North Carolina, along with a meeting of the program technical guidance committee. The test involved runway edge lighting used by helicopters. A paratroop jump zone marking and paratroop jump test originally planned for this demonstration were scrubbed due to inclement weather. This has been rescheduled for the Spring of 1983 at Ft. Benning, Georgia.

Oak Ridge National Laboratory. A test was held at ORNL using a remote site set up as an air strip (grass) for helicopter landing and combat maneuvers with the radioluminescent tritium lights and night vision equipment. The aircraft, a Huey helicopter, was supplied by the North Carolina National Guard. Edge lighting and infrared glide slope markers were deployed for use in training with night vision equipment. The field will be used for prototype light testing.

3.2.1.1.7 Instrument Lighting. Work was started in FY 1982 to evaluate radioluminescent lights for cockpit instrument lighting to fulfill a need for cockpit lighting that is compatible with advanced night vision equipment. Present incandescent lighting causes the electro-optical systems of this equipment, designed to operate at very low-light levels, to fail by aperture closing when excessive light is intercepted. The radioluminescent light is compatible with the advanced night vision equipment; therefore, it may be used for cockpit lighting. While limitations of no capability for changes in intensity or on/off warning light signals are a negative factor in instrument lighting, the reliability of light sources and ease of retrofit makes the effort attractive. An assortment of instruments that represent types used on Huey helicopters was obtained from the Aircraft Depot at Tucson, Arizona. Illumination is obtained by reflection onto the dial face from light sources placed in the frame around the dial. Good visibility was obtained. Two of the instruments have been fitted with

visible light sources. Maximum instrument visibility can be obtained by back lighting the instruments. However, this would require major instrument redesign in some cases; therefore, its application would be limited and retrofit expensive.

A second area of radioluminescent light development related to instrument lighting involves external aircraft markers. Two helicopter rotor tip lights were prepared for placement on the ends of the rotor blade. These lights consisted of a donut shaped tube encased in polyester resin and attached to the rotor inspection plate. Extensive testing to determine structural integrity will be necessary before these lights can be deployed as an aircraft component. This work was started late in FY 1982 and will continue into the FY 1983 effort.

A major accomplishment in the program was achieved by R. E. Nelson and W. C. Remini in arranging for flight time support by the North Carolina Air National Guard. Since this guard unit has as its objective 100% qualified night vision operator capability, their participation is an important key in the byproduct utilization program.

3.2.1.2 Examination of WESF Cesium Capsule

A $^{137}\text{CsCl}$ Waste Encapsulation and Storage (WESF) capsule, No. C-117, was transferred from the Sandia Irradiator for Dried Sewage Solids to ORNL for an intensive examination of its physical properties. See Figure 3.2.4 for a detailed description of a WESF capsule. The capsule was loaded on September 25, 1975, and opened on September 11, 1981. The examination of the capsule and the results obtained include the following:

- (a) Composite in-cell photographs of both the inner and outer capsules were obtained through a Köhl-Morgan periscope (see Figures 3.2.5 through 3.2.7).
- (b) Skin temperature measurements of the inner and outer capsules were made using a chromel-alumel contact thermocouple. The results of the WESF capsule temperature measurements are:

<u>Capsule</u>	<u>Skin Temperature (°F)</u>
Outer	220
Inner	260

Figure 3.2.4 Hanford Waste Encapsulation and Storage Facility Capsule

Figure 3.2.5 In-Cell View of WESF CsCl Outer Capsule

Figure 3.2.6 In-Cell View of WESF CsCl Inner Capsule

Figure 3.2.7 In-Cell View of CsCl Salt Contained in Inner Capsule

- (c) The gas contained in both the inner and outer capsules was quantitatively recovered by penetration of the stainless steel walls with a laser beam and recovering the released gas in an evacuated vessel of known volume. The gas was then analyzed using a gas chromatograph/mass spectrometer (gc/ms). The evacuated capsules were then each backfilled with He to determine the total void volume. The results of the quantitative identification of gases are presented in Table 3.2.1. The results of the void volume determination are:

<u>Capsule</u>	<u>Void Volume (cm³)</u>
Inner	244.942
Outer	134.056

Table 3.2.1. Quantitative Identification of Gases Contained in Inner and Outer Capsules

<u>Compound</u>	<u>Outer Capsule Gas Volume (cm³ at STP)</u>	<u>Inner Capsule Gas Volume (cm³ at STP)</u>
H ₂	1.358	0.023
He	<0.046	25.927
CH ₄	0.033	0.006
H ₂ O	<0.001	0.007
N ₂ +CO	58.104	11.419
O ₂	0.020	0.002
Ar	5.380	18.777
CO ₂	0.722	0.028

- (d) The mass of cesium chloride recovered from the inner capsule was 2.69kg (5.93 lb). This represents a 2.63% loss, as the reported CsCl loading was 2.76kg (6.09 lb). Results from the cesium specific activity analysis of CsCl salt are:

<u>Isotope</u>	<u>Ci/gram</u>
¹³⁷ Cs	22.1 ± 0.3
¹³⁴ Cs	0.33 ± 0.003

The results of the cesium isotopic abundance analysis of the CsCl salt are:

<u>Isotope</u>	<u>Fraction</u>
Cesium - 133	0.57
Cesium - 134	~10 ppm
Cesium - 135	0.10
Cesium-137 + Barium	0.33

(e) Metallographic examination of the capsule wall surfaces and weldments revealed a corrosion depth of 1 to 2 mils (~0.05 mm) in the wall sections examined. No corrosion of the weldments was detected.

(f) Five groups of tensile specimens were machined from selected sections of the stainless steel inner and outer capsules.

These included:

1. Top section of the inner capsule which contained the CsCl salt-void interface.
2. Bottom section of the inner capsule which was full of CsCl salt.
3. Section of the outer capsule.
4. Section of the new-design capsule* which was turned down to the same wall thickness as capsule C-117.
5. Section of the new-design capsule utilizing full thickness of material.

Tensile testing of the specimens yielded smooth creep rate versus force curves indicating a ductile metal. This is the expected curve for 316L stainless steel. The specimens were cut with the tensile axis parallel to the tube axis. Specimen loading was by pins through the grips and specimen ends. The tensile tests were conducted on an Instron tensile test machine at a cross head speed of 0.05 in. per minute. All tests were conducted at room temperature (24°C). The cross-sectional dimensions were measured before the test with a micrometer and post-test measurements were made using a toolmaker's microscope.

The tensile test results for the specimens removed from capsule C-117 are presented in Table 3.2.2 and for specimens removed from the unirradiated tubing in Table 3.2.3. In addition to the strength and ductility values presented, engineering and true fracture stresses and uniform elongation were calculated. The fracture stresses are not reported because

*The new-design capsule wall thickness is 0.1375 in. compared to 0.97 in. wall thickness for capsule C-117.

specimen necking after fracture initiation produced errors in area measurements at fracture. All measured uniform elongations were $80 \pm 2\%$ of the total elongations.

Table 3.2.2 Tensile Properties of WESF CsCl Capsule C-117 Specimens

<u>Strength (ksi)*</u>				
Thickness (in.)	Yield 0.2% offset	Ultimate	Elongation in 1 in. (%)	Reduction of Area (%)
<u>Top Inner Capsule (Above CsCl-Void Interface)</u>				
0.097 ± 0.006	37.6 ± 3.0	79.1 ± 6.0	60.2 ± 4.8	74 ± 6
<u>Bottom Inner Capsule (Below CsCl-Void Interface)</u>				
0.096 ± 0.003	47.3 ± 4.8	82.8 ± 4.7	49.8 ± 5.3	75 ± 2
<u>Outer Capsule</u>				
0.112 ± 0.001	38.3 ± 1.0	81.3 ± 1.9	61.5 ± 6.6	68 ± 4

*Results reported are average values $\pm$ standard deviation.

Table 3.2.3 Tensile Properties of Unirradiated Specimens

<u>Strength (ksi)*</u>				
Thickness (in.)	Yield 0.2% offset	Ultimate	Elongation in 1 in. (%)	Reduction of Area (%)
<u>Group 4 (see section (f))</u>				
0.094 ± 0.001	44.6 ± 4.3	95.1 ± 4.1	54.5 ± 5.3	66 ± 7
<u>Group 5 (see section (f))</u>				
0.144 ± 0.001	44.4 ± 4.6	93.4 ± 4.0	52.0 ± 1.4	62 ± 5

*Results are average values $\pm$ standard deviation.

The longitudinal tensile properties of the type 316L stainless steel inner and outer capsule, No. C-117, met the requirements of the ASTM Standards A-240, A-312, A-376, and A-511. Somewhat higher strength and decreased ductility were observed for the type 316L stainless steel in

contact with the salt. One possible explanation for these differences may be the diffusion of some elemental impurities of the salt into the stainless steel resulting in an "intergranular" attack. Further examinations are required to confirm this hypothesis.

3.2.1.3 Cesium-137 Low-Solubility Compounds

Assaying of the fully-loaded ^{137}Cs aluminosilicate pellets leachant samples was completed. The total elapsed time the pellet specimens were leached are:

Pellet Specimen	Total Elapsed Leaching Period (days)
Quasi-static, ^a sectioned, fully-loaded ^{137}Cs aluminosilicate pellets	245
Static, ^b unsectioned, fully-loaded ^{137}Cs aluminosilicate pellets	206

^a Soxhlet extractor leaching.

^b Immersion in distilled water.

The leach rates calculated from the assay results for the aluminosilicate pellets are presented in Figures 3.2.8, 3.2.9, 3.2.10, 3.2.11, 3.2.12, and 3.2.13.

The results presented in Figures 3.2.8 and 3.2.9 for pellets F1 and F2 show that the leach rates have remained essentially constant, within experimental error, after the first week of leaching. This indicates that steady-state has been achieved and the leaching of the cesium ion out of the pollucite crystal lattice is diffusion controlled (see reference 1).

Examination of the pellet leach rates for the quasi-static leaching reveals that each pellet achieved a steady-state equilibrium with respect to cesium ion leachability through the pollucite crystal lattice structure. Pellets 79CsP-4, 79CsP-12, and 79CsP-7 achieved approximately equal leach rates (see Figures 3.2.10, 3.2.11, and 3.2.13). However, the leach rate attained by pellet 79CsP-11 was an order of magnitude lower than those attained by the remaining three test pellets.

A review of the pellet fabrication history provides information which seems to explain the anomaly. The pellets were fabricated under similar

Figure 3.2.8 Aluminosilicate Pellet F1 Cesium Ion Leach Rate vs Time

Figure 3.2.9 Aluminosilicate Pellet F2 Cesium Ion Leach Rate vs Time

Figure 3.2.10 Aluminosilicate Pellet 79CsP-4 Soxhlet Cesium
Ion Leach Rate vs Time

Figure 3.2.11 Aluminosilicate Pellet 79CsP-12 Soxhlet Cesium
Ion Leach Rate vs Time

Figure 3.2.12 Aluminosilicate Pellet 79CsP-11 Soxhlet Cesium
Ion Leach Rate vs Time

Figure 3.2.13 Aluminosilicate Pellet 79CsP-7 Soxhlet Cesium
Ion Leach Rate vs Time

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conditions of temperature and pressure except for one variable, curing time duration. The curing time (i.e., the length of time the power is applied to the pellet after ram travel has ceased) for pellet 79CsP-11 was 90 minutes as compared to 20, 45, and 10 minutes, respectively for the remaining pellets. This indicates a minimum time, t (where $45 < t \leq 90$) is required in which to assure that all of the cesium is incorporated into the pollucite crystal lattice. Otherwise, the cesium can be leached from the pellet at a rate approximately 10X greater (see reference 1).

3.3 PUBLICATIONS

F. J. Schultz, J. A. Tompkins, K. W. Haff, F. N. Case, Preparations and Characterizations of Cesium-137 Aluminosilicate Pellets for Radioactive Source Applications, ORNL-5775 (July 1981).

F. N. Case, K. W. Haff, J. A. Tompkins, F. J. Schultz, W. C. Remini, Radioactive Powered Light Sources, ESL-TR-82-12 (in publication); Engineering and Services Laboratory, Air Force Engineering and Services Center, Tyndall Air Force Base, Florida.

F. N. Case and K. W. Haff, Krypton-85 Powered Lights for Airfield Applications, ESL-TR-80-55 (November 1981); Engineering and Services Laboratory, Air Force Engineering and Services Center, Tyndall Air Force Base, Florida.

3.4 REFERENCES

1. F. J. Schultz, J. A. Tompkins, K. W. Haff, F. N. Case, Preparation and Characterization of Cesium-137 Aluminosilicate Pellets for Radioactive Source Applications, ORNL-5775 (July 1981).
2. K. W. Haff, F. N. Case, F. J. Schultz, J. A. Tompkins, Testing of Tritium-Powered Runway Distance and Taxiway Markers, ESL-TR-81-45 (August 1981); Engineering and Services Laboratory, Air Force Engineering and Services Center, Tyndall Air Force Base, Florida.

4. SURPLUS FACILITIES MANAGEMENT PROGRAM

J. H. Coobs

4.1 INTRODUCTION

This is the first of two programs that are concerned with the management of surplus facilities. The facilities in this program are those related to defense activities, which include the ORNL Graphite Reactor, about 20 waste collection and storage tanks that have been retired from service, abandoned waste transfer lines, the Fission Product Development Laboratory (FPDL), and a number of other process and waste facilities. The program is an integral part of the Surplus Facilities Management Program, which is a national program administered for DOE by the Richland Operations Office. Objectives of the program are to provide for surveillance and maintenance of surplus radioactively contaminated DOE facilities awaiting decommissioning in order to assure safe conditions and to implement a structured decommissioning program to accomplish disposition of all surplus DOE-owned facilities. The decommissioning operations are supported by financial estimating, planning and scheduling, engineering, and technology development activities.

4.2 PROGRESS

Technical progress for each of the tasks of this program for fiscal year 1982 follows.

4.2.1 Defense Surplus Facilities Surveillance and Maintenance (R. W. Schaich, R. V. McCord, J. F. Alexander, and F. J. Peretz)

This task includes survey and inspection of surplus facilities on a regular basis, collection and analysis of samples from waste streams, replacement of exhaust air filters on a regular basis, and maintenance of facilities as indicated by procedures and results of inspections.

Routine monitoring of containment and replacement of exhaust filters was performed at all facilities. Monitoring of waste systems and storage cells and maintenance of manipulators was carried out at FPDL. The demineralizer system was regenerated at the Graphite Reactor and the exhaust stack was cleaned and inspected. The storage canal at the Metal Recovery Facility was monitored and analyzed to detect increases in contaminants, and badly deteriorated roof hatches were repaired. A consolidated report on all phases of the radiological characterization of the Metal Recovery Facility was completed and issued, and a companion report giving details of the analysis and evaluation of subsurface soil cores near the facility was completed. Development of special equipment and procedures for survey and sampling of abandoned waste tanks was begun.

4.2.2 Curium Facility Decommissioning (R. W. Schaich and F. J. Peretz)

The Curium Source Fabrication Facility (CSFF) was designed and constructed in the early 1960s to prepare and encapsulate multigram quantities of curium-242 and curium-244 for the U.S. Atomic Energy Commission's Systems for Nuclear Auxiliary Propulsion (SNAP) program. At the termination of the SNAP program, the facility operated as a curium-244 fuel development facility until 1979. The facility, shown in

Fig. 4.1, contains four manipulator operated cells surrounded by water-filled tanks for neutron shielding and a concrete- and steel-shielded cell that was used for final source encapsulation. Decontamination of the facility was begun in January 1981.

At the time decontamination was begun, the selected mode for decommissioning the facility was dismantlement of the water-shielded cells and decontamination and removal of excess equipment from the other cells and the cell access area. This complete decommissioning program was estimated to cost about \$2.7 million and would have made the vacated facility available for further modification and reuse. On the other hand, early in FY 1982, a more immediate use for the facility became evident because of the impending transfer to ORNL in FY 1983 of the uranium-234 recovery program. This program would occupy the Alpha Handling Facility in Bldg. 3038, which is presently utilized for americium operations. The proposal was made and preliminary approval obtained for transfer of the americium-241 program to the Curium Facility, contingent upon completion of the decontamination work at the Curium Facility.

Decontamination efforts during FY 1981 concentrated on high-pressure spraying with aqueous detergent solution to remove insoluble particles of $^{244}\text{Cm}_2\text{O}_3$ from cell surfaces. This method removed much contamination but was not effective in reducing alpha smear levels much below 1×10^6 dpm/100 cm^2 .

The next stage in the decontamination cycle was accomplished early in FY 1982 by applying a Turco-4501 solution to all in-cell surface

Fig. 4.1. Curium Source Fabrication Facility.

areas, followed by a warm oxalic acid solution rinse to remove oil, grease, and rust that had accumulated on the non-stainless steel equipment. Following the solution decontamination cycle, all in-cell surface areas were wiped with wet cotton swabs containing a high percentage of solid sodium bisulfate (Sani-Flush cleanser). Test wipes were removed from the cell system and were found to be alpha contaminated to greater than 1.0 MBq ($>6 \times 10^7$ dpm). This decontamination procedure was repeated until the analytical data indicated a general contamination level of less than 2.0 KBq ($<1 \times 10^5$ dpm) alpha per wipe. The use of swabs was then discontinued and all accessible surface areas inside Cell 3 were covered with a strippable paint (ALARA Decon Yellow). A plastic tent with an airlock entry was constructed around the Cell 3 access plate, and operating personnel protected with plastic airline suits removed the cell access plate under controlled conditions. All vessels, piping, and associated equipment were coated with strippable paint, transferred to Cell 3, and placed in stainless steel drums for storage as TRU waste.

Final decontamination of the in-cell surfaces of the water shielded cells was accomplished by the wet wipes saturated with cleanser, which reduced the in-cell surface area contamination to less than 1.0 KBq ($<6 \times 10^4$ dpm) alpha per wipe. A final application of strippable paint to the in-cell surfaces will reduce the contamination level to an acceptable tolerance for in-cell maintenance and repairs that are planned for FY 1983.

The Beta/Gamma Manipulator Cell (Cell 5) was contaminated with particles containing alpha- and beta-emitting radionuclides (<2 KBq per wipe)

due to a variety of processes that were performed in this cell over the past 20 years. The decontamination techniques used on Cell 5 were identical to those previously reported. After the contamination level had been reduced to less than 1 KBq ($<6 \times 10^4$ dpm) alpha or beta on a smear, operating personnel entered the cell in plastic airline suits, removed excess equipment, and scrubbed the in-cell surface areas by hand. An aqueous detergent solution and cleanser-pads were used until the smear level was reduced to less than 100 Bq (<6000 dpm) alpha per wipe. The surface was then painted with an epoxy paint (Colma Sol) and the smear level was reduced to ORNL requirements (1.0 Bq or 60 dpm alpha) for the future operations planned for this cell.

Decontamination of the outer surfaces of the water shielding tanks and the cell access areas is in progress and will be completed early in FY 1983. The extended-reach manipulators that were removed from the cells were decontaminated to acceptable levels, repaired, and stored for future program use. This decontamination project will be completed at an estimated total cost of ~\$700,000 and will make available for alternate use a facility that would cost several million dollars to replace.

4.2.3 ILW Transfer Line Decommissioning (A. A. Walls, W. G. Tatum, and S. P. du Mont)

The scope of this task involves removal of approximately 210 m (700 ft) of the abandoned 2-in. ILW pipeline from the White Oak Creek floodplain, removal of the transfer line from the Old Hydrofracture Facility to the New Hydrofracture Facility, and entombment of two sections of the ILW line where leaks occurred plus sealing to divert surface water and prevent erosion.

The transfer line from the Old Hydrofracture Facility to the New Hydrofracture Facility was removed in December 1981. This 800-ft segment of heavy-wall pipe had been used only for a test injection at the new facility and was not heavily contaminated. Sections of this pipeline were disconnected and recovered during removal and have been stored for future use.

The schedule for removal of the ILW transfer line from the floodplain was advanced to the fourth quarter of FY 1982 to avoid interference with construction activities scheduled for September and October 1982 on the new weir and channel for Monitoring Station No. 3. Excavation was begun in July, and the two parallel lines were exposed, disconnected, and drained. The stainless steel line was lifted, starting at the end near the old hydrofracture facility, cut into 9-ft sections, and packaged for disposal as low-level waste. Two 20-ft sections of the original cast iron pipe that were still connected to the stainless steel line at the east were also removed and cut into 30-in. sections for packaging into drums and storage as TRU waste. This was necessary because analyses of samples of the cast iron and mild steel pipe showed that it had adsorbed sufficient quantities of transuranic nuclides to be classified as TRU waste whereas the stainless steel pipe was much less contaminated and could be disposed of as low-level waste.

The abandoned carbon steel line which paralleled the stainless line was lifted in 20-ft sections and then cut into 30-in. lengths for disposal as TRU waste. Both of the carbon steel pipe and stainless steel pipe sections which crossed White Oak Creek and extended west were

lifted as individual long sections from the creek bed and relocated for sectioning. Three 50-gal drums were used to store the 300 ft of carbon steel pipe. All of the stainless steel pipe removed was packaged into 9-ft sections and disposed of as low-level waste. The operations during excavation and removal of the pipelines are shown in Fig. 4.2.

Radiation readings on the pipe were consistent with previous surveys, i.e., <30 mr/h on the exterior of the pipe and 10-12 R/h on the interior of the pipe. Some holes were found in the carbon steel line as a result of corrosion, as expected. This was especially the case for the portion of the line that was exhumed from the creek bed.

The packaged sections of pipe were removed to storage or disposal, and the excavations were closed to complete this portion of the project.

4.3 PUBLICATIONS

F. J. Peretz and J. F. Alexander, Results of the Radiological Characterization of Building 3505, UCND Report X-OE-190 (September 1982).

4.4 REFERENCES

None.

5. HIGH-LEVEL WASTE MANAGEMENT PROGRAM

W. W. Pitt

5.1 INTRODUCTION

The Defense High-Level Waste Management Program for FY 1982 consisted of only one task, "Concrete Grout Bulk Immobilization Study." The objective of this task was to provide an option for the management of existing waste at the Hanford site by:

- (a) the development of tailored cement-based waste forms for Hanford wastes,
- (b) evaluation of product performance and performance criteria, and
- (c) providing assistance to Hanford on the selection and development of transport and emplacement technologies.

ORNL has a long history of successful laboratory and engineering development of cement-based grouts used as hosts for the immobilization of radioactive wastes. Examples of radioactive wastes fixed in tailored cement based hosts include: ion-exchange media, evaporation bottoms, filter media, waste sludges, waste slag, incinerator ash, waste calcines, shredded metal, and heavy waste oils. Through the use of additives and the application of advanced processing technology, the performance of cement based waste forms have been tailored for site specific cases. In addition, over 2 million gallons of intermediate level waste solutions and sludges have been successfully disposed of via an engineering scale grout facility, providing over 15 years of operating experience.

5.2 PROGRESS

Technical progress for the three subtasks for FY 1982 follows.

5.2.1 Concrete Grout Bulk Immobilization Study (E. W. McDaniel, L. R. Dole, G. A. West, and R. L. Hickey)

During FY 1982 the following tasks were addressed: (1) grout mix development of simulated Hanford wastes, (2) leach studies on simulated Redox waste, and (3) a preliminary study of leaching performance criteria.

5.2.1.1. Grout Mix Development of Simulated Hanford Wastes

Simulated supernate and sludge wastes were supplied by Rockwell Hanford for grout studies. The wastes included: (1) simulated Redox wastes, (2) three simulated Purex wastes, including non-processed, neutralized current acid waste (NPN-CAW), washed sludge and digested sludge, and (3) Zirflex/cladding wastes including $Mg(OH)_2$ supernate and sludge and NaOH supernate and sludge.

Grout formulation studies were made for each type of waste. The initial formulations were based on those used at ORNL to dispose of waste solutions and sludges by the hydrofracture process. These basic formulations were modified to accommodate the particular waste stream chemistries and to optimize waste loading, leachability and physical properties of the resulting grout. Generally grouts were made by mixing wastes and water with a dry solids blend of cement, fly ash and clays (Indian Red and/or Attapulgate-150) in a ratio of one liter (waste/water) per 0.8 Kg of dry solids. Admixtures (<1 wt %) were added when needed as set regulators or waste reducers. The resulting waste-loaded grouts typically contain 10-16 wt % cement, <16 wt % fly ash, and

<10 wt % clays with the remainder being wastes and water. Raw materials available locally in the Pacific Northwest region were used whenever possible for the grout mixes. These included Type I-II LA cement from the Oregon Portland Cement Company, Durkee, Oregon, and Pacific Northwest pozzolan ASTM Class F fly ash from a Centralia, Washington power plant. Reference grouts were also made from standard ORNL materials including Portland Type I cement, Kingston TVA fly ash, and Ottawa graded sand.

Since the leach performance and physical properties are very sensitive to the maturity of the paste (i.e., curing time), these data must necessarily lag behind the selection of acceptable formulas. Therefore, the selection of mix formulas was based on the grouts' flow properties, which establish the formulas' compatibility with standard processing technology, such as mixing, pumping, and emplacement equipment. The standard technology has evolved from oil field grouting through the development and 20 years operation of the ORNL hydrofracture in situ solidification process. Pertinent flow properties include fluid consistency index, flow behavior index, density, critical velocity, pumping rate for critical velocity, fractional pressure drop and hydrolic horsepower.

5.2.1.1.1. Grouts Prepared with Simulated Redox Waste. Formulation studies for the fixation of simulated Redox waste in a cement-based grout showed that pumpable grouts could be made with locally available materials. Two water reducing agents, Plastiment® from Sika Chemical Corp. and REAX-LP® of Westvaco, Inc., were used in these tests. Rheology of the grouts made from locally available materials and standard ORNL materials indicate that raw materials have a major effect on

grout rheology and that Plastiment® is an effective admixture in the formulas using Northwestern materials.

5.2.1.1.2. Grouts Prepared with Simulated Purex Waste. Formulation studies for the fixation of simulated Purex waste in cement-based grouts and the resulting grouts' flow properties, were investigated for three types of waste: (1) non-processed, neutralized current acid waste (NPN-CAW) solution, (2) simulated Purex sludge that had been washed and centrifuged, (3) simulated Purex sludge that had been digested at 92°C for 72 h, then washed and centrifuged.

Grouts were made with NPN-CAW waste using local Northwestern and the standard ORNL cements and fly-ashes. Three admixtures, Plastiment®, D-65® from Dowell, Inc., and CFR-1® from Halliburton Services, Inc., were used for each cement mixture. These grouts were plastic and fluid. Their rheological data demonstrated that these grouts' flow properties are also very sensitive to the presence of admixtures. Grouts were also prepared with NPN-CAW waste that had been digested at 92°C for 72 h, using Attapulgate-150 drilling clay as a suspender. This grout was also fluid, plastic and suitable for deep geologic disposal.

Grouts were prepared with two types of simulated Purex sludge: (1) washed and centrifuged, and (2) digested at 92°C for 72 h, washed and centrifuged. The grouts made with the digested sludge had a lower viscosity, lower pressure drop, and higher density than the undigested sludge. However, the grout with digested sludge exhibited a much higher 120 min. gel strength, which could create problems, such as, fouling of pipelines and pumps. However, such problems are avoided by proper formulation design and equipment selection (i.e., the pump has the capacity

to break maximum gel strength and the formulation is designed to be < this maximum).

Based on these results and ORNL's experience with grouting, the following conclusions can be drawn about Purex wastes:

1. Grouts prepared with non-processed, neutralized current acid waste solutions and an ORNL hydrofracture grout dry solids blend behave like hydrofracture grouts.
2. Grouts prepared with simulated Purex sludge and a dry solids blend, used at ORNL to prepare sludge-slurry grouts, behave like ORNL sludge-slurry grouts.
3. Admixtures are very effective in controlling grout flow properties.
4. Grouts prepared with digested and nondigested sludges, both washed and centrifuged, are different in character.

5.2.1.1.3 Ziflex Wastes/Cladding Removal Wastes (CRW). Ziflex decladding waste results from dissolution of the Zircaloy-2 cladding from N-reactor fuels in a boiling ammonium fluoride - ammonium nitrate solution. This solution is neutralized by addition of NaOH resulting in a slurry consisting primarily of $ZrO_2 \cdot xH_2O$ sludge and a supernatant containing NaF. An alternative method is being evaluated at Rockwell Hanford Operations (RHO), where the decladding waste is neutralized by addition of magnesium hydroxide rather than NaOH. The resulting neutralized waste would consist of a mixture of $ZrO_2 \cdot xH_2O$, MgF_2 , and $Mg(OH)_2$. This supernate should be nearly free of soluble ions.

A major area of concern of fixing simulated Zirflex waste in a cement based matrix is the effect of fluorides on set-time of cement, since NaF is a commonly used set retarder. Using $Mg(OH)_2$ instead of NaOH to neutralize the Zr-cladding leachate (strike) eliminates this problem. The $Mg(OH)_2$ precipitates soluble fluoride ions; thus reducing or eliminating the fluoride.

Thus, disposal of four types of waste were investigated:

- (1) supernate of $Mg(OH)_2$ strike of Zr-cladding leachate,
- (2) solids produced as a result of $Mg(OH)_2$ strike of Zr-cladding leachate,
- (3) supernate of NaOH strike of Zr-cladding leachate, and
- (4) solids produced as a result of NaOH strike of Zr-cladding leachate.

Scoping tests made by mixing each simulated waste with the same dry solids blend showed that hydrofracture compatible grouts can be prepared with both $Mg(OH)_2$ strike supernate and solids. The supernate produced a high phase separation (i.e., H_2O collecting on top of grout), when mixed with a dry solids blend. The phase separation can be reduced or eliminated by increasing the dry solids.

The qualitative set (hardening) data are as follows:

1. The grout made with supernate of $Mg(OH)_2$ strike was rigid to the point that the gel could not be broken after 72 h. This is 24 h longer than a normal hydrofracture grout with the same mix ratio of waste to dry-solids and retarded with CRF-1[®] set regulator. This grout had a strong odor of ammonia.
2. The grout made with NaOH strike supernate showed no setting after 72 h.

3. The grout made with $Mg(OH)_2$ strike solids started into initial set after 72 h, but this set could be broken. This grout had a strong ammonia odor, also.
4. The grout made with NaOH strike solids showed no setting at the end of 72 h.

Based on these scoping tests, the following conclusions can be drawn:

1. A grout suitable for a variety of disposal options can be prepared with $Mg(OH)_2$ strike supernate.
2. Grout prepared with NaOH strike supernate exhibits a longer set-time than is desirable. Additional development work will be required to shorten the set-time.
3. Grout can be made that contains $Mg(OH)_2$ strike solids, but additional development work will be required.
4. Grout made with NaOH strike solids is similar to that made from NaOH strike supernate.

5.2.1.2 Leach Studies on Simulated RHO Redox Waste

Preliminary results are presented for static leach tests performed according to MCC-1 specifications on a grout formula designated RHO-4 using a basalt-bentonite leachant. Composition of RHO-4 is listed below:

<u>Material</u>	<u>Composition (wt %)</u>
Portland Type I Cement	26.45
Kingston Fly Ash	5.75
Ottawa Graded Sand	3.07
Indian Red Clay	3.07
Sludge (dry wt)	17.95
D-65® Water Reducer	0.59
Water	43.13

The leachant was prepared by contacting deionized water with a Basalt-Bentonite mixture. The Basalt was obtained from the Rockwell Hanford Operations Energy Systems Group. The Bentonite was a beneficiated Wyoming Sodium Bentonite (trade name Thixo-gel #2).

The leach tests were performed at 40°C with leachate samples taken at 3, 7, 14 and 28 days. In general, the data was consistent and showed no unexpected results. In any static leach test (such as MCC-1), the leachate will become saturated with time. In this experiment the leachate pH, conductivity, and concentrations of aluminum, sodium, and potassium were constant after 3-7 days. The leachate had not quite been saturated with calcium in 28 days, although the solution appeared to be rapidly approaching it. Significantly, the parabolic relationship observed between the calcium leachate concentration and leach time is a strong indication that the calcium release is due to diffusion of Ca(OH)_2 , and it does not indicate matrix dissolution.

Three methods of determining release rates for selected elements were compared. Release rates were calculated as follows:

$$NL_i = \frac{M_i}{f_i \cdot SA \cdot t} \quad (1)$$

$$L_i = \frac{M_i}{SA \cdot t} \quad \text{and} \quad (2)$$

$$F_i = \frac{M_i}{G_i \cdot t} \cdot 100\% \quad (3)$$

where

NL_i = Normalized leach rate of component i [$\text{g/m}^2 \cdot \text{day}$]

M_i = Mass of element i in leachate [g],

f_i = Mass fraction of element i in solid sample,

- SA = Solid sample surface area [m^2],
 t = Leach time [days],
 L = Leach rate of components [$g/m^2 \cdot day$],
 F_i = Fractional release rate of component i [%/day], and
 G_i = Total mass of component i in solid sample [g].

In general, the calculated release rates showed no surprises. As expected with a static leach test, the average release rates decreased with leach time. The alkali and alkaline earth elements showed a markedly higher release rate than the heavier matrix elements. For example, 70% of the sodium was released in a 3 day period. It should be noted that these specimens were not allowed sufficient curing time to develop their potential physical properties. This resulted in higher release rates, than would be expected of mature specimens that have been cured 90-128 days. Subsequent leach studies will be performed using mature specimens.

5.2.1.3 Preliminary Study of Leaching Performance Criteria

5.2.1.3.1 Background. In order to estimate the potential performance required of a grout waste host at the Hanford site, two scenarios were selected: (1) solidification in single-walled tanks, and (2) bulk solidification in the basalt repository. Assuming the 10^{-5} annual fractional release limit of the proposed NRC 10 CFR 60, the necessary leaching properties were back calculated with the second conservative assumption that takes no credit for the tank-and-liner and tubing-and-backfill for cases 1 and 2, respectively.

5.2.1.3.2 Methods and Results. Figures 5.1 and 5.2 were calculated, using the Joy-Godbee model:

$$F \cdot V/SA = (R \cdot D)^{1/2} [(t + 1/2R) \operatorname{erf}(R \cdot t)^{1/2} + (t/\pi R)^{1/2} \exp(-R \cdot t) + 1] \quad (4)$$

where

F = cumulative fraction leached at time $[t]$,

V = volume of sample $[cm^3]$,

SA = surface area of sample $[cm^2]$,

$R = U^2/4D$, moving boundary constant $[s^{-1}]$,

D = apparent diffusion coefficient $[cm^2/s]$,

t = time $[s]$,

U = moving boundary velocity or degradation rate $[cm/s]$, and

$\operatorname{erf} u$ = error function = $2/\pi^{1/2} \int_0^u e^{-z^2} dz$.

This model has been shown to effectively scale the surface (SA) to volume (V) effects from small (15 cm^3) to large (100 L) samples.

Therefore, these results indicate that a smaller SA/V ratio of large monoliths allows the use of grouts with a higher leach rate and still meet the intent of 10 CFR 60.

Figure 5.1 shows the effective diffusion constant (D_e) and the apparent boundary velocity R required for a concrete monolith in the shape of a Hanford AX tank. For grouts with $D_e < 10^{-13} cm^2/s$ and $R < 10^{-10} s^{-1}$, the unprotected waste solid would meet the intent of 10 CFR 60. In the case of a 200 L drum (Figure 5.2), much lower leach rates are required. At this time, the configuration of the waste solid monolith is not known for the basalt repository scenario. For now, we assume that its SA/V-ratio will be approximately that of a 200 L drum and that a $D_e < 10^{-16}$ and

ORNL DWG 82-508

Fig. 5.1. Parameters necessary to obtain the maximum NRC release fraction of 1×10^{-5} in 1 year from Hanford AX tank (dimensions 75 ft diam x 36.5 ft high).

ORNL DWG 81-455R

Fig. 5.2. Parameters necessary to obtain the maximum NRC release fraction of 1×10^{-5} in 1 year from standard 200-L drum.

Fig. 4.2. Excavation for Removal of ILW Transfer Lines from White Oak Creek and Floodplain. View looking west toward White Oak Creek from the Old Hydrofracture Facility. Excavation in foreground has been partially closed, and removal of pipes is in progress near the creek.

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a $R < 10^{-12}$ will be required. ORNL has shown that the present hydrofracture grouts and autoclaved concretes satisfy these performance requirements.

5.3 PUBLICATIONS

None.

5.4 REFERENCES

None.

7.1 DOE LOW-LEVEL WASTE MANAGEMENT PROGRAM

L. J. Mezga

7.1.1 - INTRODUCTION

In early FY 1979 Oak Ridge National Laboratory was assigned the associate lead contractor role in the DOE Low-Level Waste Management Program (LLWMP). This role is carried out under the overall management of the Program Manager at DOE's Idaho Operations Office, the lead contractor, (EG&G Idaho) at the Idaho National Engineering Laboratory, and DOE's Oak Ridge Operations Office.

The overall objective of the LLWMP is to develop and promote environmentally sound techniques for safely and efficiently managing the disposal of all types of low-level wastes (LLW).

LLW results from such activities as energy production, manufacturing, and the use of radioisotopes in medical and research activities. Specific program technology objectives include the development of waste treatment and packaging technology; development of improved shallow land burial technology; establishment of corrective measures for stabilizing and improving performance of existing burial grounds; identification and evaluation of alternative methods for storage and disposal of LLW that provide greater confinement than shallow land burial; and development of guidelines or standards for waste characterization, disposal, and monitoring.

The Oak Ridge National Laboratory (ORNL) in its role as associate lead contractor of the DOE LLWMP has responsibility for the management of program-funded technology development activities. In this role with general guidance provided by DOE and the lead contractor (EG&G Idaho), the ORNL program office is charged with the responsibility to

- (1) develop program plans for the major technology areas, (2) recommend allocations for the program resources, (3) review the technology development tasks to ensure that program objectives are being met, and (4) to assist the lead contractor in coordinating the DOE LLWMP with other on-going U. S. and foreign waste technology programs. Although the ORNL office generally assists the lead laboratory in management of the total program, our emphasis is on management of R&D for development of basic technology and to assess concepts for alternative systems of processing and disposal of LLW.

7.1.2 - PROGRESS

Technical progress for each of the tasks of this program for FY 1982 follows.

7.1.2.1 Low-Level Waste Management (L. J. Mezga, R. B. Pitts, M. S. Moran, W. H. Pechin, L. E. Stratton, and D. J. Wilkes)

7.1.2.1.1 Technical Planning

Details for the technology development portions of the LLWM Five-Year Program Plan were developed in conjunction with EG&G Idaho. The actual issuance of the detailed overall Program Plan for FY 1982 was

carried out by EG&G Idaho. The following reviews of technology status were prepared by this office in support of program planning and control:

- Revision of Position Paper: "Evaluation of the Need for Greater Confinement than Shallow Land Burial of Low-Level Wastes" (L. J. Mezga),
- Final Report: "Review of Liquid Waste Treatment Processes Developed by the DOE LLWMP," ORNL/NFW 82/7 (J. E. Vath), and
- Final Report: "Verification and Validation Protocols for Low-Level Waste Disposal Site Performance Models" (M. S. Moran).

In order to insure that program outputs are of consistently high quality and delivered on schedule, the following program guidance was developed for technology development tasks:

- Guidance for the Preparation of Experimental Test Plans,
- Technical Guidance for the Preparation of FY 1983 Current Year Work Plans, and
- Technical Guidance for the Preparation of FY 1984 Field Task Proposals/Agreements.

7.1.2.1.2 Resource Allocation

The major effort during FY 1982 focused on the development and prioritization of the FY 1982, 1983, and 1984 budgets. These

activities included: (1) preparation of FTR/A and budget call technical guidance, (2) evaluation and revision of FY 1982, 1983, and 1984 budgets, (3) evaluation and revision of the FY 1982, 1983, and 1984 Current Year Work Plans, and (4) prioritization of the FY 1982, 1983, and 1984 budgets. These were iterative processes based on changes in the proposed budgets for these years and were made more complex because of the problems associated with operating under a continuing resolution.

7.1.2.1.3 Program Review

Program review activities during FY 1982 included: (1) conducting site visits to existing contractor facilities, (2) briefing DOE-HQ and field offices, and (3) conducting the Third- and Fourth-Annual Information Meetings. Site visits were conducted to review and discuss current fiscal year activities and to develop the elements of the Current Year Work Plan for the coming fiscal year. LLWMP staff provided guidance to the various contractors relative to program goals and technical needs, anticipated funding levels, and program organization during these visits. The information gained during this exchange was used in evaluating and prioritizing the FY 1982, 1983 and 1984 budgets.

Information exchange and program coordination within the LLWMP occurred during the Third- and Fourth-Annual Information Meetings held in New Orleans, Louisiana on November 4-6, 1981, and Denver, Colorado on August 31 and September 1-2, 1982. The proceedings of the third annual meeting were published as ORNL/NFW-81/34. The proceedings of the fourth annual meeting are in press. During these meetings, the

various contractors reported on the status of projects funded during the current fiscal years (FY 1981 and 1982). A series of workshops were held during the third annual meeting to obtain input from the contractors regarding research needs and program approach. The fourth annual meeting consisted almost entirely of small group, technical sessions to maximize information exchange and discussion among the participants. The audience for this meeting was expanded to include representatives of USNRC, USEPA, USGS, several states, and private consultants and facility operators. Contractors from all four federal agencies presented reports on their work allowing for better coordination and understanding of the federal program.

7.1.2.1.4 Coordination With Other Waste Technology Programs

Information exchange and program coordination with other DOE and federal waste programs continued. In addition to the major effort on coordination with other federal waste programs that occurred at the Third- and Fourth-Annual DOE LLWP Program Participants' Meetings, the ORNL LLWM program office participated in an interagency meeting on the application of models to low-level waste disposal held in Germantown, Maryland, June 15, 1982. This meeting brought together representatives from the U. S. Department of Energy, U. S. Environmental Protection Agency, U. S. Geologic Survey, and U. S. Nuclear Regulatory Commission.

Coordination with other defense waste programs was maintained through participation in the Headquarters Field Coordination Meeting. In addition, a memorandum of agreement was developed between the DOE LLWM and TRU Programs identifying areas of programmatic interaction

related to greater confinement disposal. To that end, TRU Program representatives participated in the Fourth Annual Participants Meeting.

7.1.2.2 Site Participation

This task provided support for the Third and Fourth Annual DOE LLWMP Participants Meetings. These activities are discussed in section 7.1.2.1.3 Program Reviews.

7.1.2.3 Corrective Measures Handbook (L. E. Stratton)

7.1.2.3.1 Corrective Measures Criteria

Draft Corrective Measures Criteria were prepared and submitted to EG&G Idaho in July 1982, two months ahead of schedule, for use as part of DOE Order 5820. This completed ORNL's effort on the criteria. The Corrective Measures Criteria provide standards for addressing problems encountered with the shallow land burial of low-level radioactive waste by identifying threshold levels which require corrective measures to improve disposal site performance. In developing the criteria, an initial working draft was prepared and a review committee was formed with representation from the DOE, the states, and the commercial sector. The committee recommendations were used as foundation principles for developing the criteria.

The draft corrective measures criteria provide for correction of conditions that could jeopardize performance objectives of a site. Conditions covered in the criteria are: (1) wind and water erosion,

ponding or flooding, (2) accumulation of water in trenches, (3) plant or animal intrusion, and (4) chemical or biological reactions.

7.1.2.3.2 Corrective Measures Handbook

The Corrective Measures Handbook describes the experience of shallow land burial site operators with the implementation of corrective measures to eliminate conditions that result, or could result, in the site's failure to meet its stated performance objectives. The report provides operators with documentation of previous efforts to correct conditions common to several SLB disposal sites that result in inadequate site performance. As such, the report is a "state-of-the-art" document and will be the reference point for the final Corrective Measures Handbook.

The report is based on a literature review and on visits to six DOE and five commercial disposal sites. Experience indicates that the most effective corrective actions are those developed based on an understanding of the site conditions which caused the problems. Accordingly, the report is organized by causative factors (i.e., conditions that have resulted in problems). These factors include: (1) unstable trench cover, (2) permeable trench cover, (3) subsidence, (4) groundwater entering trenches, (5) intrusion by deep-rooted plants, (6) intrusion by burrowing animals, and (7) chemical and physical conditions in the trench. In discussion of each, the condition is briefly introduced and is followed by an itemization of actions to correct the condition.

A peer review committee was created with representation from both DOE and commercial disposal facility operators. The committee reviewed the report and their recommendations were incorporated. The current practices corrective measures report titled, "Previous Experience and Related Research and Development in Applying Corrective Measures at the Major Low-Level Radioactive Waste Disposal Sites," was completed in September 1982 as scheduled.

7.1.3 PUBLICATIONS

D. E. Large, R. S. Lowrie, L. E. Stratton, and D. G. Jacobs (editors), 1981. Proceedings of the Third Annual Information Meeting, DOE Low-Level Waste Management Program, November 4-6, 1981, New Orleans, Louisiana, ORNL/NFW-81/34, 586 pages.

D. E. Large, L. E. Stratton, L. J. Mezga, R. R. Rose, Compilers, 1982. Proceedings of the Fourth Annual Participants' Information Meeting, DOE Low-Level Waste Management Program, August 31, September 1-2, 1982, Denver, Colorado, ORNL/NFW-82/18 (in press).

7.1.4 REFERENCES

None.

7.2 ORNL LOW-LEVEL WASTE PROGRAM

N. H. CUTSHALL

7.2.1 INTRODUCTION

Projects, addressing the generic aspects of radioactive solid waste disposal, are carried out in the ORNL Low-Level Waste Research and Development Program. These include the evaluation and demonstration of improved burial procedures for future and existing disposal sites and the development and demonstration of corrective measures and strategies for site stabilization and closure. The federal regulation code (10 CFR 61) requires that all future commercial disposal sites be managed to meet a performance objective of no off-site groundwater contamination which would result in a radiological dose above certain regulatory limits. To meet this requirement, certain sites may require engineered barriers, such as lined or grouted trenches, as improvements to the current practice of dump-and-fill. Thus, a significant effort is under way to demonstrate and evaluate grouting and lining techniques. Even with good site planning and advanced burial techniques, the proposed regulations also require contingency plans for corrective actions should site performance be compromised by groundwater contamination. Techniques for groundwater contamination management, particularly for the more mobile radionuclides contained in the waste, need to be developed and demonstrated before generic recommendations for them can be advocated. Techniques under evaluation include a passive groundwater diversion system, remedial trench grouting, and chemical treatment of trenches and groundwater seeps with caustic soda and soda ash. The final phase

in the management of a waste disposal site is its stabilization and closure. After closure, a site is abandoned to the processes of geological weathering and vegetational succession through which period the site must still meet certain performance objectives. An evaluation of the interaction of weathering processes and vegetational succession with management techniques for final site stabilization is being pursued. Possible strategies to minimize these processes from compromising site performance need to be identified so that the consequences of various strategies for site stabilization and closure can be predicted.

7.2.2 PROGRESS

Technical progress for each task of this program for fiscal year 1982 follows

7.2.2.1 ORNL Corrective Measures (B. P. Spalding, M. G. Browman, E. C. Davis, L. K. Hyder, D. S. Marshall, I. L. Munro)

Shallow land burial (SLB) has been and will continue to be the dominant disposal method for most low-level radioactive solid waste. SLB trenches usually perform well in retaining radionuclides and in preventing their migration to uncontrolled areas. Occasionally, trenches do leak radionuclides to the extent that corrective measures seem warranted. Problems with SLB including the infiltration of surface runoff into burial trenches, groundwater intrusion, trench subsidence, soil erosion, trench intrusion by plants and animals, and radionuclide migration, have been identified as potentially or actually leading to situations requiring corrective action. A

hypothetical trench with all of these identified problems is depicted in Figure 7.2.1. The seasonal fluctuation of the water table leads to periods when the buried waste is bathed by groundwater. Additional inundation is facilitated by direct infiltration of precipitation and the catching of any surface runoff due to partial subsidence of the trench backfill. Points of rapid trench subsidence or cover collapse can effectively function as drains for surface runoff leading to an accumulation of water which results in a perched water table within the trench. Obviously, the control and diversion of surface runoff can alleviate many of these inundation problems. Trench cover sealing with bentonite or plastic sheeting has been advocated to relieve the situation of direct infiltration of precipitation (7.2.1). However, trenches like the one depicted in Figure 7.2.1 often are situated in areas where the water table is maintained by lateral groundwater recharge rather than, or in addition to, direct infiltration precipitation. Passive groundwater interception and diversion drains provide a means to relieve such situations.

Such engineered drainage alterations, however, can become economically prohibitive for large regions containing many trenches, particularly when only one or a few trenches are the source of the problem. In these situations, hydrologic isolation of individual trenches can be addressed better by waste grouting. The injection of grout throughout the void space of a burial trench, which upon grout set becomes impermeable to water, can effectively isolate

Fig. 7.2.1. Problems encountered in the shallow land burial of low-level waste to be addressed by corrective measures technology.

buried waste from further groundwater intrusion. In addition, since trench void spaces become filled, subsidence can also be effectively controlled.

Mobility of radionuclides in low-level waste is the source of its potential hazard. One radionuclide, ^{90}Sr , is particularly troublesome for SLB not only in a generic hazard evaluation because of its high radiotoxicity and its high mobility in soil but also at specific sites such as ORNL's waste disposal areas (7.2.2). Fixation or immobilization of ^{90}Sr in soil, in and around a burial trench, has been pursued both in laboratory and field studies using caustic soda and soda ash chemical treatment. Such chemical techniques would be most applicable to situations where either groundwater diversion or grouting would be expensive. Treatments would also be applicable to groundwater seep contamination where an immediate abatement of radionuclide concentrations may be required. Treatment at a source trench may require a considerable interval before effects are observed at a distant contaminated seep.

The ORNL corrective measures technology task is divided into four subtasks: a passive groundwater diversion drainage demonstration, a trench grouting demonstration, a trench chemical treatment demonstration, and a groundwater seep treatment demonstration.

7.2.2.1.1 Passive Drainage Demonstration

This subtask addresses the situation where groundwater intrusion of a group of trenches is maintained by lateral recharge rather than by direct infiltration of precipitation. By constructing a groundwater intercepting and diverting drain between the area to be

ORNL-DWG 79-20420AR2

Fig. 7.2.2. Location of proposed french drain for groundwater diversion demonstration Solid Waste Storage Area 6 at ORNL.

protected and its recharge area, the groundwater recharge can be eliminated. For purposes of a demonstration of this technology, a subsection of solid waste storage area (SWSA) 6 at ORNL, containing 49 burial trenches, was selected as a site for construction of a french drain (Figure 7.2.2). The suite of 49 trenches, to be protected by the drain, was surface sealed with a soil-bentonite mixture in November, 1976; however, seasonally standing water still exists in many of the trenches (7.2.3). Whether or not the bentonite seal is working in this area cannot be determined from observations of the water table alone; in addition to any infiltration of precipitation, allowed by a poorly performing seal, the water table in this area is maintained by lateral groundwater recharge from areas not under the seal. Interestingly, based on the surface topographic contours and water table elevation data in the area (7.2.4), a traditional hydrologic analysis would indicate that the recharge area would be the shaded area in Figure 7.2.2. However, the area immediately east of these 49 trenches is probably more important as a recharge zone because it lies along the line of joint fractures in the area (7.2.5) which function as preferred groundwater flow paths. Additional experimental determinations of the groundwater velocities and directions along the line of the proposed drain are being collected. Regardless of the relative contributions of these two possible recharge areas, the 49 trench area will be protected by the surrounding drain. Test borings along the proposed drain have been completed and no hard rock was encountered within the 7.6 meter maximum depth of the drain. Construction will begin in the near future

and, in the pre-construction interval, water table elevations will continue to be monitored to add to the existing five-year record of the area (7.2.4).

7.2.2.1.2 Trench Grouting Demonstration

This subtask addresses the situation where a particular trench, rather than a group of trenches, is functioning as a source of radionuclide groundwater contamination and, therefore, compromising the performance of a site. By the injections of a low viscosity and non-particulate grout into the void space of a problem trench, the buried waste can be hydrologically isolated after the grout sets to its impermeable state. Two basic problems are faced by the potential grout user for any application as a corrective measure for low-level radioactive solid waste. Firstly, of the many grout types and formulations presently available (7.2.6), a selection must be made using criteria including a low-grout viscosity similar to that of water, a set time controllable from minutes to days, an inherently low permeability after set, durability to resist the physiochemical and microbial stresses in its applied environment, low toxic hazard, and, of course, low cost. Any grout selection will, of necessity, represent a compromise among these properties. Based on existing data (7.2.6), five grout formulations were selected for laboratory testing; the fundamental property for this laboratory evaluation was the ability of the grout to reduce the hydraulic conductivity of several soils and waste types. Both 10% polyacrylamide and 15% sodium silicate performed well and were selected for field demonstration.

The second problem to be faced by any grout user for such corrective measures application is how to achieve a distribution of the grout within the target trench. Since both the acrylamide and sodium silicate grout formulations are low viscosity and non-particulate, they would flow within trench voids as facilely as intruding groundwater. Therefore, grout injection could be effected with a ~~minimum~~ number of injection wells; because the hydraulic conductivity of a trench's waste/backfill is much greater than the surrounding unperturbed soil formation, trenches can be facilely filled with fluids. To test this critical assumption for waste grouting application, three disposal trenches were constructed in the suspect waste landfill area of SWSA 6. Trenches were filled with suspect solid waste and backfilled with soil as SLB is currently practiced at ORNL. A grid of nine wells was positioned in each trench and each trench was surrounded by four additional wells within the unperturbed formation. By measuring the hydraulic conductivity of the material surrounding each of these wells through constant-head pump-in tests, grouting effectiveness can be determined by comparing before and after grouting measurements. The grid of wells, in addition to measuring the waste/backfill and soil formation conductivity, provides a means to determine the areal extent of grout penetration.

Both silicate and polyacrylamide grouts were readily injected into the trenches and were observed in all monitoring wells. The hydraulic conductivities of all wells are dramatically reduced with more than 80% becoming effectively impermeable. Continuing monitoring

of the permeability of each grouted trench will establish the longevity of their effectiveness.

7.2.2.1.3 Chemical Trench Treatment Demonstration

Treatment of trenches with caustic soda and soda ash can lead to a significant fixation or immobilization of ^{90}Sr which is likely the most compromising radionuclide to the performance of a SLB site. Laboratory studies, employing both test tube batch and soil columns, have indicated that $\text{Ca } (^{90}\text{Sr}) \text{CO}_3$ is an excellent phase to form in soil in-situ by the addition of soda ash and/or caustic soda (7.2.7). A number of concerns remained about the applicability of this technique to acidic soils which are inherently low in available Ca from which CaCO_3 would form on the addition of soda ash. Several acidic soils were treated with soda ash and supplemental Ca and Ba to supply adequate amounts of alkaline earths to effect a precipitation of CaCO_3 or BaCO_3 , respectively. Soil columns were treated with ^{85}Sr and chemicals as described previously (7.2.7) and eluted with dilute CaCl_2 . Supplemental Ca or Ba did effect a greater fixation of ^{85}Sr than soda ash alone. Thus a two-step system for the chemical fixation of radiostrontium in strongly acidic soils would seem quite feasible and considerably broadens the range of soils to which the technique is applicable.

The field demonstration of the trench treatment with caustic soda and soda ash, reported last year (7.2.8), has continued in its monitoring phases. Little change in the trends of the observations reported then has occurred to the present. Hydrologic characterization of the trench

environs has been attempted with some water pump-in tests. The resulting hydrologic characteristics will be used to evaluate the chemical behavior of this trench through time and will be used to formulate a model of the trench environs from which conclusions about trench performance beyond the period of experimental observation can be based.

7.2.2.1.4 Groundwater Seep Treatment Demonstration

This subtask addresses the situation where a groundwater seep is discharging radioactivity into surface water. Such a situation may require an immediate corrective action depending on the level of radioactivity involved. Source treatments, like the three techniques discussed above, could not result in an immediate response at a seep discharge because the contamination plume, leading to the seep, could take several years to flush. Thus, even interim techniques to address such situations have considerable merit. A seep on the periphery of SWSA 5 was selected for a field demonstration and a weir, stage height recorder, and automatic water sampler were installed at the site. After about a year of pre-treatment monitoring, 360 kg of soda ash was buried at seven locations immediately upslope from the seep and the response of ^{90}Sr and seep water quality were observed. Following soda ash burial, pH and electrical conductivity of the seep water increased markedly, while the hardness and ^{90}Sr levels decreased concomitantly. An in-situ softening of this seep water, with the coprecipitation of ^{90}Sr on calcite, was brought about by this treatment. Average water quality parameters are presented in Table 7.2.1 the periods 0-300 days

before, 0-90 days after, and 91-310 days after soda ash burial. An average 80% reduction in ^{90}Sr concentration was maintained for the 90-day period following soda ash burial which was followed by a slow recovery to near pretreatment levels over the next year. Thus, the technique has merit as an interim corrective measure, which could be repeated as often as necessary, while more permanent corrective measures were carried out at the source. The major reason for its temporariness is the depletion of the soda ash via dissolution, neutralization, and calcite precipitation in the groundwater flow path. Since the incoming ^{90}Sr in groundwater was not stopped through a source treatment, once the soda ash was depleted, there was no mechanism for continued radiostrontium fixation.

7.2.2.2 Shallow Land Burial Technology Development. (N. D. Vaughan, G. T. Yeh, F. G. Taylor, N. H. Cutshall, I. L. Larsen, L. H. Stinton, O. M. Sealand)

The purpose of this task is to adapt, develop and demonstrate technical procedures that support the shallow land burial of low-level radioactive waste. Three subtasks have been pursued during the past year: 1) Advanced Burial Procedures, 2) Site Stabilization and Closure and 3) Site Selection Procedures Evaluation.

7.2.2.2.1 Advanced Burial Procedures.

The Engineered Test Facility (ETF), is a five-year field demonstration project of improved shallow land burial procedures. Site characterization and waste emplacement have been completed and monitoring phases begun.

Table 7.2.1 Average chemical characteristics of seep water in the periods before, 90-days after, and greater than 90-days after soda ash treatment of a groundwater seep.

Characteristic	Units	Period*		
		0-300 days before	0-90 days after	91-310 days after
^{90}Sr	Bq/mL	7.37a	1.36b	5.34c
Hardness	mgCaCO ₃ /L	390a	83b	290c
Conductivity	μS/cm	638a	9000b	662a
pH	-log[H ⁺]	7.4a	9.5b	7.6a
Number of Samples	—	63	93	106

* Means followed by the same letter are not significantly different at the 5% level.

7.2.2.2.1.1 Site Characterization. The ETF is located within the Copper Creek thrust block of the Valley and Ridge Province of east Tennessee and is underlain by strata of the Middle to Lake Cambrian Conasauga Group. The Maryville Limestone formation, which is composed of ribbon-bedded and interclastic limestones and dark grey shales and mudstones, comprises the bedrock immediately beneath the site. The bedrock and residuum structure are characterized by anticlinal folds with numerous joints and fractures, some of which are filled with calcite. Seismic and electrical resistivity methods were useful in characterizing the thickness of residuum and presence of structural features. Soils are illitic and range from podzolic to lithosols to alluvial in the vicinity of the ETF, but the original soil solum was removed in 1975 when the mixed hardwood forest was cleared and the site was planted in grasses. The remaining residuum consists of acidic soil aggregate (pH 4.4 to 5.6) and extensively weathered siltstone and sandstone which exhibit the original rock structure. Mean annual precipitation at the site is 1500 mm, although during the initial study period (10-1-80 to 9-30-81) the annual total was 939 mm. Runoff was estimated to be approximately 50% of total precipitation based on observations at two Parshall flumes installed at the site. Storm runoff is quite responsive to rainfall and the lag time between peak rainfall and runoff is less than 15 minutes during winter storms. Water quality of surface runoff is as expected for a mixed carbonate and siliceous terrain; except, there is tritium migration from an adjacent disposal area

present in the flume to the east of the study area. Tracer studies of the groundwater system, together with pump tests, have shown the important influence of fractures and joints on groundwater movement at the site. The results to date suggest that groundwater flow has two distinct components; one associated with fracture flow and the other with intergranular flow (7.2.9).

7.2.2.2.1.2 Waste Emplacement. Waste emplacement consisted of waste-type selection, identification of specific liner and grout treatments, and actual emplacement. Compacted, baled low-level radioactive waste was chosen because of the quantity of waste needed for the triplicate demonstration-sized trenches ($\sim 27\text{m}^3$) for each treatment (control, grout and liner). These trenches were constructed in a Latin-square array within the space enclosed by the characterization wells. The bales were dumped as opposed to stacked into the trenches to simulate the voids normally found in low-level disposal trenches.

A 1.1 mm reinforced Hypalon material was selected as the liner. It was placed into the trenches prior to the bales and then sealed shut after the trench was full. The grout used was composed of 0.45 kg bentonite and 2.75 kg Type 1 cement for every 3.78 liters of slurry and was introduced after the waste was emplaced. These treatments are designed to retard water movement into the waste thus reducing subsequent radionuclide migration from the trenches and reducing trench subsidence. Each trench had a unique chemical tracer placed in it at time of treatment for monitoring purposes.

7.2.2.2.1.3 Monitoring. For the monitoring phase 24 wells were drilled with each trench surrounded by at least 4 wells. Samples are taken from the monitoring and characterization wells on a biweekly schedule to be analyzed for chemical or radionuclide contamination. Results indicate that tracer breakthrough has not occurred in the first 4 months after closure.

An automated data acquisition system has been developed for monitoring groundwater levels at the ETF site. The system employs a battery-operated data logging instrument that digitizes potentiometer output from water-level monitors in the 15 ETF wells, and records the digitized data on cassette tape at approximately 14-minute intervals. At one to two week intervals, the cassette is removed from the logger, and its contents are transferred onto a disk file on ORNL's PDP-10 computer. The raw data are preprocessed on the PDP-10 to convert from reverse octal to decimal millivolts. An interactive program prompts the user for the date, time, and calibration information associated with a given tape, and creates a raw data file for transfer to, and permanent storage, on the IBM Mass Storage Subsystem. A SAS (Statistical Analysis System) program that runs on the IBM 3033 converts the raw millivolt data to water-level elevations, associates these elevations with appropriate date-time information, and performs several checks for consistency of calibration and timing. The new data are added to a data base (stored as a SAS data set) which contains all previously collected data in a form readily accessible for plotting and further data analysis, including model validation.

7.2.2.2.1.4 Burial Site Transport Models. Numerous models using finite difference and finite element approaches for simulating the water flow and contaminant transport in either saturated-unsaturated porous media or in completely saturated aquifers have been reported in the literature (7.2.10, 7.2.11). However, none of the computational codes have been designed to simultaneously handle the transient sources/sinks, Dirichlet (prescribed head or concentration), Neumann (prescribed gradient of head or concentration), Cauchy (prescribed flux of water and contaminant), and variable boundary conditions (also confining leaky aquifers in the case of completely saturated system). In real world, problems such as occur in disposal sites, recharge/pumping, are more likely to be distributed over the region of interest and to be time dependent. The boundary values, with rare exception, vary both with time and along a Dirichlet boundary. The prescribed gradients are both temporally and spatially distributed over the Neumann surfaces. The total fluxes vary with time and along Cauchy surfaces. Furthermore, in many occasions, the boundary conditions cannot be predetermined but change with the passage of time and with field conditions. For prototype applications, a computational model must have the capability to deal with these aspects efficiently and simultaneously. Four computer codes, FEWA (7.2.12), FEMA (7.2.13), FEMWATER-1 (7.2.14), and FEMWASTE-1 (7.2.15) have been developed and implemented to achieve these objectives in the past year. FEWA and FEMA are intended for application to completely saturated aquifers over horizontal areas. This pair of models can be applied to large-scale as well as small area problems. On the other hand, FEMWATER-1

and FEMWASTE-1 are intended for applications to combined saturated-unsaturated porous media on a vertical cross-sectional area. They are more applicable to problems on a scale of hundreds of meters. They can be applied to larger-scale problems although computer storage requirements and central processing unit (CPU) time will be large. These four computer codes are developed and modified based on FEMWATER (7.2.16) and FEMWASTE (7.2.17). The modifications will consider: (1) space and time-dependent sources/sinks representing recharges and pumping; (2) highly transient boundary values on Dirichlet segments; (3) time-dependent, distributed gradients over Neumann boundaries; (4) highly time dependent total fluxes on Cauchy boundaries; (5) addition of off-diagonal hydraulic conductivity to the water-flow equation; (6) options of under- and over-relaxation techniques in iteration solutions; and (7) automatic resetting of time step size when boundary conditions or sources/sinks change abruptly. In addition, the input structure is completely reformulated so that practitioners can use them easily. Modifications 1 through 4 make the model applicable to realistic situations when internal sources/sinks and boundary conditions are changing with time either gradually or abruptly and are distributed over space. Modification 5 enables the application of these models to cases of geologic media consisting of several formations, each is nonhomogeneous and anisotropic. Modification 6 extends model applications to wider ranges of parameters when they make the numerical solutions fluctuate or converge too slowly. Modification 7 makes the schedule of time step size easy upon change of conditions. In addition,

point-wise iterative computation is built in FEWA and FEMA as an alternative to the direct solution. This makes FEWA and FEMA applicable to large-scale problems because the requirement of large computer storage due to large band width of the matrix is relaxed.

7.2.2.2.2 Site Stabilization and Closure

Two factors that are critical to the long-term success of shallow land burial site closure are soil erosion and vegetation control.

7.2.2.2.2.1 Soil Erosion Measurement. Post-closure soil erosion may affect shallow land burial sites by reducing the integrity of trench covers, allowing infiltration into the trench and, in extreme cases, exposing the waste. Agricultural erosion work has previously led to development of the "universal soil loss equation" which estimates erosion rates based on precipitation, land surface slope, soil erodibility, etc. The USDA computer model, CREAMS, represents an advanced implementation of the universal soil loss equation. Direct field measurements of erosion have been made to confirm the validity of the universal soil loss equation for relatively flat mid-western terrain. This subtask will determine if the validation can be generalized to a wider set of terrains. It is conducted in cooperation with Dr. G. A. Neilsen and Mr. O. Arnalds at Montana State University.

Direct analysis of soil movement in the field is slow and expensive. Therefore, indirect trace methods have been developed to allow more rapid estimation of erosion and deposition. Cesium-137 was used as a soil tracer in erosion studies by Rogowski and Tamura (7.2.18). Their techniques have been adapted to ^{137}Cs from fallout atmospheric

weapons tests, refined and applied by McHenry (7.2.14), Ritchie (7.2.20) and coworkers and by Brown et al. (7.2.21, 7.2.22). These techniques will be used in variety of terrains in Montana, ranging from prairie farmland to high alpine and the results compared to estimates from the universal soil loss equation. A preliminary investigation has been completed and fallout ^{137}Cs appears to be a promising indicator of soil movement in Montana. In particular, two sets of samples have been analyzed. The first set was selected from a wind eroded field. A field which had been plowed just prior to a major historical windstorm was selected. Plow layer samples taken from the field itself were depleted in ^{137}Cs reflecting the loss of soil that had occurred. Cesium-137 inventories in paired samples taken to the windward and leeward sides of windbreaks and fencerows downwind from the eroded field also showed the effects of soil movement. Windward inventories are 1.5 to 2 times the leeward inventories showing the accumulation of soil that occurred.

The other set of samples were taken in a small watershed. Vertical profiles from four sites were determined (Table 7.2.2). A regular gradient in ^{137}Cs inventories is apparent. In the upper- and lower-slope areas which are subject to erosion, ^{137}Cs inventories are markedly lower than the pond at the bottom of the watershed. Thus, it appears that the results are qualitatively consistent with expectation. Approximately 200 samples are now being analyzed for ^{137}Cs , and the results will be used to make quantitative estimates of soil removal and deposition.

Table 7.2.2. Montana watershed fallout ^{137}Cs Distribution.

Cesium-137 and soil are depleted from the upper slope and deposited in the pond.

Site	Slope	Segment Thickness (inches)	^{137}Cs Concentration pCi/g Segment		
			Upper	Center	Lower
Upper slope	5-7	6	.17	.08	Not sampled
Lower slope	1-3	6	.59	.13	Not sampled
Grass near pond	flat	5	.84	.48	.07
Pond, center	flat	5	.72	.65	.53

7.2.2.2.2 Vegetation Studies in Humid Site Stabilization and

Closure. Long-term stabilization and minimization of maintenance requirements are desirable components of closure for shallow land burial sites. State-of-the-art protocols utilizing grass as cover to inhibit invasion of undesirable species are labor-intensive and not suitable for the long term. A more desirable vegetative cover would employ phases other than pioneer stages (grasses) and would persist for many years with only periodic maintenance. Considering the diverse floristic composition of the humid eastern United States, no single taxon is likely to provide an ideal cover. Succession models can identify potential cover types (grasslands, thickets, forests) that can be manipulated (maintained) to prolong dominance.

Preliminary work with simulation models indicates plantations of certain early successional arboreal species may provide suitable vegetation cover while requiring infrequent, labor-intense manipulation. Species such as red maple or white ash have proper rooting characteristics, sprout readily when harvested, and can regenerate and maintain growth, through coppicing (stump sprouting) on a 20-year cycle, while providing high leaf areas suitable for relatively high evapotranspiration. Current work indicates highly variable growth under different moisture and nutrient regimes. The use of solely successional models, however, oversimplifies the temporal character of a cover in simulated studies. Addition of moisture and nutrient sub-routines in model studies, superimposed upon manipulative practices,

will provide design criteria in cover recommendations throughout the southeast.

From an initial selection process, red maple and white ash represent choices for plot studies to begin next year. Efforts will be combined with site-specific research to evaluate the species ability to cover an area, interaction with invading (competing) species, and persistence without intense maintenance.

7.2.2.2.3 Site Selection Procedures Evaluation

During FY 1982 work was initiated on the Site Selection Procedures Evaluation Subtask. The subtask goal was to demonstrate the application of site selection criteria developed by DOE to a specific region - the Oak Ridge Reservation. Generally, the site selection process consists of two stages - site screening and site characterization. Site screening activities are based on reconnaissance level data such as published or unpublished reports, existing records, and brief field surveys. Site screening culminates in the identification of a preferred site. Site characterization entails the evaluation of the preferred site to confirm that the site characteristics will meet the specified performance standards. Significant site data and analysis are required to confirm site performance. Generic site selection procedures were identified covering the entire site selection process. The availability of information at the Oak Ridge site required that the specific site application concentrate on the site screening stage.

Initially the work was designed to use a computer code for site selection evaluation. However, upon reviewing existing computer codes for selecting a low-level waste disposal site, none were directly applicable at the site screening stage. Several models were available for site characterization; but, if these were forced into service for site screening, the data assumptions and model manipulations required would leave little credibility in the results. A generic site selection methodology was developed to provide a framework for applying the DOE criteria to the Oak Ridge Reservation.

Upon reviewing existing site selection criteria and procedures, a generic methodology was documented. The methodology was generally adapted from state-of-the-art siting strategies and tailored for the specific needs of low-level waste disposal and the DOE site selection criteria. In general, the site screening involved defining the region of interest, identifying candidate areas and sites, and ranking candidate sites to identify the preferred site. The site selection strategy is summarized in Figure 7.2.3. The detailed logic for each activity was also developed. The logic includes provisions for iterations and paths of recourse within the methodology in case the preferred site is found to be unacceptable.

Work was also initiated on applying the methodology to the Oak Ridge site and comparing the results to previous siting studies completed on the reservation. These activities will be completed during the first quarter of FY 1983.

Fig. 7.2.3. Logic flow for shallow land burial site selection procedures. a) Siting screening, b) characterization.

7.2.2.3 US/FRG Cooperative Program. (L. R. Dole, G. C. Rogers, and J. H. Kessler)

This task is directed toward a cooperative research and development program and information exchange on low-level waste disposal with the Federal Republic of Germany concerning in-situ solidification of low-level waste for disposal in salt cavities or other geological formations. Research has been initiated to determine the feasibility and applicability of this procedure as an alternative technique to shallow land burial. The tasks include: (1) consultation and information exchange with the FRG, (2) data assessment for applicability to shallow land burial, and (3) establishment of a cooperative research and development program.

Funding for this task ended after 3 months of FY 1982 and the work was terminated.

7.2.2.4 Decontamination Effectiveness. (G. L. Copeland)

Work on this program ceased in January when the program was canceled. At this point, information gathering for the literature survey was complete, and preparation of the report had started. Another aspect of the program was to provide more detailed analysis of ingots and slag produced during the demonstration melting at the Paducah Gaseous Diffusion Plant of 100 tons of contaminated steel. About 15 tons of steel were melted before suspected liner erosion caused melting to be stopped. These 15 tons of steel were cleared to about 1 ppm uranium. This program was canceled during the delay in melting, and the planned doping of scrap with Cs and Sr was not accomplished.

7.2.2.5 Shallow Land Burial Handbook

Additional disposal facilities are needed to accommodate the low-level radioactive wastes being produced by both commercial and defense activities. One of the purposes of DOE's Low-Level Waste Management Program is to "develop the technology and documentation required to open a shallow land burial site." The *Shallow Land Burial Handbook* provides support for this goal by transferring state-of-the-art information about shallow land burial technology to the user community. The handbook gives developers and state or local planners an overview of and initial guidance for undertaking the development of phases and activities required for establishing new facilities for near-surface disposal of low-level radioactive waste. These activities, carried out from the first feasibility studies through closure and postclosure, include defining and establishing the scope of the problem, establishing objectives, compiling information, considering alternatives, and establishing and implementing plans.

This document went through several revisions and review cycles during FY 1982. Reviews of the final product indicate an acceptable document which should be useful to the U.S. Department of Energy and others in the government and private sector.

7.2.2.6 Low-Level Waste Treatment Handbook

The objective of the *Low-Level Waste Treatment Handbook* is to present an overview of current practices related to the segregation, classification, volume reduction, solidification, handling, packaging, and transportation of low-level radioactive waste for disposal in a shallow land burial facility. The handbook is intended to serve as a guide to individuals who have no previous knowledge in handling low-level wastes and who need an introduction to the available technology and regulations.

The growing concern with low-level waste disposal has focused attention on methods for reducing the volumes being generated. One method of volume minimization is to exercise better administrative controls to limit the production of waste, such as contaminated dry trash. Volume minimization can also be achieved by more efficient waste treatment options; however, maximum volume reduction will dictate equipment modifications involving capital investment and licensing considerations. The commercial introduction of more sophisticated systems such as incineration for combustible wastes can significantly reduce the low-level waste volumes for disposal. Since waste characteristics and volumetric burial charges are unique to each generator, cost-effective waste treatment procedures will have to be developed on a site-specific basis.

The handbook describes low-level radioactive waste treatment from source to ultimate disposal, including an introduction to the waste treatment problem, the sources and characteristics of low-level wastes; treatment and volume reduction techniques for both liquid and dry wastes, including ion exchange, evaporation, filtration, reverse osmosis, compac-

tion, incineration, and dehydration; low-level waste solidification techniques, including a discussion of sorbents, cement, bitumen, and organic polymers; and packaging and transport of low-level waste to the disposal facility. To reflect commercially available technology, Gilbert Associates, Inc., was subcontracted to address the commercial status of low-level radioactive waste treatment and handling procedures.

7.2.3 PUBLICATIONS

T. E. Cerling, and B. P. Spalding. "Distribution and Relationship of Radionuclides to Streambed Gravels in a Small Watershed," Environ. Geol., (in press).

T. E. Cerling and R. R. Turner. 1982. "Formation of Freshwater Fe-Mn Coatings on Gravel and the Behavior of ^{60}Co , ^{90}Sr and ^{137}Cs in a Small Watershed," Geochim. et Cosmochim. Acta, 46:1333-1343.

N. H. Cutshall, N. D. Vaughan, C. S. Haase, C. R. Olsen, and D. D. Huff. "Hydrologic and Geologic Aspects of Low-Level Waste Site Management," Proc. International Conference on Radioactive Waste Management, Winnipeg, Manitoba, Canada, Sept. 12-15, 1982, (in press).

B. P. Spalding and D.D. Huff. 1981. "Remedial Action Technology for Humid Sites," pp. 437-466, Proceedings of the Third Annual Participants Information Meeting, DOE Low-Level Waste Management Program, ONL/NFW-81/34. Oak Ridge National Laboratory, Oak Ridge, TN

B. P. Spalding and W. A. Hoffman, Jr. 1982. Soil Radiochromatography using ^{22}Na , ^{85}Sr , and ^{58}Co , ORNL/TM-3138. 44 pp.

B. P. Spalding. 1982. "Research Efforts on Environmental Transport of Hazardous Materials at Oak Ridge National Laboratory," Trans. Amer. Nuc. Soc., 41:66.

B. P. Spalding, M. G. Browman, and E. C. Davis, "Corrective Measures Technology for Humid Sites," Proc. of the Fourth Annual Participants Information Meeting, DOE Low-Level Waste Management Program, ONL/NFW-82, (in press).

N. D. Vaughan and N. H. Cutshall. "SLB Technology-Humid". Proceedings of the Fourth Annual Participants Information Meeting, DOE Low-Level Waste Management Program, ONL/NFW-82, in press.

N. D. Vaughan, C. S. Haase, D. D. Huff, S. Y. Lee, and E. C. Walls, Field Demonstration of Improved Shallow Land Burial Practices for Low-Level Radioactive Solid Wastes: Preliminary Site Characterization and Progress Report, ORNL/TM-8477, in press.

G. T. Yeh, "Computation of Darcian Velocity and Mass Balance in the Finite Element Modeling of Groundwater Flow," Water Resource Research, Vol. 12, No. 5, 1529-1534.

G. T. Yeh and T. Tamura. "Geohydrochemical Consideration in Land Disposals of Low-Level Wastes," Nuclear Sciences and Engineering, Volume 82, No. 2, 206-219.

G. T. Yeh and R. J. Luxmoore. "A Multi-Dimensional Model for Simulating Moisture and Thermal Transport in Unsaturated Aquifer Systems," Modeling and Simulations, Vol. 12, No. 1, p. 61-66, Instrument Society of America Research Triangle Park, N.C.

G. T. Yeh and R. J. Luxmoore, "Chemical Transport in Macropore-Mesopore Media Under Partially Saturated Conditions," Symposium on Unsaturated Flow and Transport Modeling, Seattle, Washington, March 23-24, 1982, p. 267-282, Nuclear Regulatory Commission, Washington, D.C.

G. T. Yeh, CHNHYD: A Channel Hydrodynamic Model for Simulating Flows and Water Surface Elevations in Stream/River Network, ORNL-5701, ESD Publication No. 1686, Oak Ridge National Laboratory, Oak Ridge, Tennessee, January 1982, pp. 123.

G. T. Yeh and R. H. Strand. FECWATER: User's Manual of a Finite Element Model for Simulating Water Flow through Saturated-Unsaturated Porous Media, ORNL/TM-7316, ESD Publication No. 1523, Oak Ridge National Laboratory, Oak Ridge, TN.

G. T. Yeh, "Simulations of Flows and Water Depth in Dendritic River System," International J. Numerical Methods in Fluids.

G. T. Yeh and R. J. Luxmoore. "Modeling Moisture and Thermal Transport in Unsaturated Porous Media," J. Hydrology.

G. T. Yeh and R. H. Strand. FECWASTE: User's Manual of a Finite Element Code for Simulating WASTE Transport Through Saturated-Unsaturated Porous Media, ONL/TM-8104, Oak Ridge National Laboratory, Oak Ridge, TN.

G. T. Yeh, CHNTRN: A Channel Transport Model for Simulating Sediment and Chemical Distribution in a Stream/River Network, ORNL-5882, Oak Ridge, TN.

G. T. Yeh and R. J. Luxmoore, MATTUM: A Multi-Dimensional Model for Simulating Moisture and Thermal Transport in Unsaturated Porous Media, ORNL-5888, Oak Ridge National Laboratory, Oak Ridge, TN..

7.2.4 REFERENCES

1. J. O. Duguid, Annual Progress Report on Burial Ground Studies at Oak Ridge National Laboratory: Period Ending September 30, 1975 ORNL-5141. 54 p. 1976.
2. J. O. Duguid, 1975. Status Report on Radioactivity Movement from Burial Grounds in Melton and Bethel Valleys. ORNL-5017. 66 p.
3. H. S. Arora, D. D. Huff, D. S. Ward, and O. M. Sealand, 1981. An Assessment of a Bentonite Seal on Groundwater Storage in Underlying Waste Disposal Trenches at ORNL. ORNL/TM-7416. 52 p.
4. D. A. Webster, J. S. Beatty, P. M. Benjamin, and W. M. Trnum, 1980. Water Level Data for Wells in Burial Ground 6, Oak Ridge National Laboratory 1975-79. USGS Open-file Report 81-57.
5. J. J. Sledz and D. D. Huff. 1981. Computer Model for Determining Fracture Porosity and Permeability in the Conasauga Group. ORNL/TM-7695. 138 p.
6. J. Herndon, and T. Lenahan, 1976. Grouting of Soils. Vol. 1. A State-of-the-Art Report. PB-259-043. NTIS. 286 p.
7. B. P. Spalding, 1981. Chemical Treatments of Soil to Decrease Radiostrontium Leachability. J. Environ. Qual. 10:42-46.
8. B. P. Spalding and D. D. Huff, 1981. Remedial Action Technology Development for Humid Sites. p. 437-446. In Proceedings of the Third Annual Information Meeting, DOE Low-Level Waste Management Program. ONL/NFW-81/34. 586 p.

9. N. D. Vaughan, C. S. Haase, D. D. Huff, S. Y. Lee and E. C. Walls,
Field Demonstration of Improved Shallow Land Burial Practices for
Low-Level Radioactive Solid Waste: Preliminary Site Character-
ization and Progress Report, ORNL/TM-8477 (in press).
10. C. A. Appel and J. D. Bredehoeft. 1976. Status of Groundwater
Modeling in the U.S. Geological Survey. Geological Survey
Circular 737, U.S.G.S., Reston, VA. pp. 9.
11. Y. Bachmat, J. Bredehoeft, B. Andrews, D. Holtz, and S. Sebastian.
1980. Groundwater Management: The Use of Numerical Models
Water Resources Monograph 5, American Geophysical Union,
Washington, D.C. pp. 127.
12. G. T. Yeh, and D. D. Huff. 1982a. FEWA: A Finite Element Model
of Water Flow through Aquifers, Oak Ridge National Laboratory,
Oak Ridge, TN, (in preparation).
13. G. T. Yeh and D. D. Huff. 1982b. FEMA: A Finite Element Model
of Material Transport through Aquifers. Oak Ridge National
Laboratory, Oak Ridge, TN (in preparation).
14. G. T. Yeh, 1982a. FEMWATER-1: A Finite Element Model of Water
Flow through Saturated-Unsaturated Porous Media - first revision.
Oak Ridge National Laboratory, Oak Ridge, TN (in preparation).
15. G. T. Yeh, 1982b. FEMWASTE-1: A Finite Element Model of Waste
Transport through Saturated-Unsaturated Porous Media - first
revision. Oak Ridge National Laboratory, Oak Ridge, TN
(in preparation).

16. G. T. Yeh and D. S. Ward. 1980. FEMWATER: A Finite-Element Model of Water Flow through Saturated-Unsaturated Porous Media. ORNL-5567, Oak Ridge National Laboratory, Oak Ridge, TN pp. 153.
17. G. T. Yeh and D. S. Ward. 1981. FEMWASTE: A Finite-Element Model of Waste Transport through Saturated-Unsaturated Porous Media. ORNL-5601, Oak Ridge National Laboratory, Oak Ridge, TN. pp. 137.
18. A. S. Rogowski and T. Tamura. 1965. Movement of ^{137}Cs by Runoff, Erosion and Infiltration on the Alluvial Captina Silt Loam. Health Physics 11:1333-1340.
19. J. R. McHenry. 1968. Use of Tracer Technique in Soil Erosion Research. Trans. Amer. Soc. Agricultural Engineers 11(5):619-625.
20. J. C. Ritchie and J. R. McHenry. 1977. The Distribution of ^{137}Cs in some Watersheds in the Eastern United States. Health Physics 32:101-105.
21. R. B. Brown, N. H. Cutshall and G. F. Kling. 1981. Agricultural Erosion Indicated by ^{137}Cs Redistribution: I. Levels and Distribution of ^{137}Cs Activity in Soils. Soil Soc. Amer. Jour. 4(6):1184-1190.
22. R. B. Brown, G. F. Kling and N. H. Cutshall. 1981. Agricultural Erosion Indicated by ^{137}Cs Redistribution: II. Estimates of Erosion Rates. Soil. Soc. Amer. Jour. 45(6):1191-1197.

8. WASTE MANAGEMENT OPERATIONS PROGRAM

J. D. Sease

8.1 INTRODUCTION

The major function of the Program is to operate the Laboratory's systems and facilities for collecting and disposing of radioactive gaseous, liquid, and solid wastes. This includes the collection and shallow land burial of about 2,000 m³ of beta-gamma contaminated waste and retrievable storage of about 60 m³ of transuranium contaminated waste annually; ion-exchange treatment and release to the environment of about 450 x 10³ m³ of slightly contaminated water; volume reduction by evaporation of about 5,000 m³ of intermediate-level liquid waste followed by hydrofracture injection of the concentrate; and scrubbing and/or filtration of the gases from radioactive operations prior to release to the atmosphere. In addition, this year disposal of about 350,000 gal of radioactive sludge from the old (no longer in service) gunite tanks began. Operations are in conformance with rules and regulations presently applicable to ORNL.

This Program is responsible for planning and for development activities for upgrading the facilities, equipment, and procedures for waste disposal; the objective is to ensure that waste management facilities and procedures at ORNL incorporate the latest technology and are consistent with DOE and other regulatory requirements. Major (line-item) new facilities are provided as well as substantial (GPP) upgrading of old facilities. Technical and engineering support is provided to handle nonroutine operating problems. An environmental research and development (R&D) group

conducts studies of our shallow land burial operations to identify site-specific problems and produce scientific basis for corrective actions -- as well as generic information of use to the National Low-Level Waste Management Program. Also, studies for volume reduction of solid waste for disposal are carried out by R&D groups supported by the Program.

Identification and design and development activities continue for a new disposal area for solid low-level waste generated at ORNL and for a central disposal facility for solid low-level radioactive waste for the three DOE plants located in Oak Ridge. Various operations are under study for disposal of the uranium-contaminated scrap metal currently stockpiled at DOE enrichment facilities. Also, work continues to develop operational tools for the DOE traffic managers in the transportation of radioactive materials and hazardous wastes.

8.2. PROGRESS

Technical progress for each of the tasks of this program for fiscal year 1982 follows.

8.2.1 Operational Planning and Development (J. H. Coobs, J. R. Gissel, R. E. Helms, S. H. Merriman, T. F. Lomenick, F. E. Harrington, T. S. Mackey, H. O. Weeren, T. F. Scanlan, T. N. Tiegs, and E. S. Bomar)

This task provides technical and engineering support and coordination services for all of the activities related to the management of solid, liquid, and gaseous radioactive wastes generated at ORNL as a result of various R&D and radioisotope production programs. The objective is to ensure that waste management procedures and facilities at ORNL incorporate the latest technology and are consistent with DOE and other regulatory requirements. A further objective of this task is to identify the need for, plan for, and bring into operation new or reno-

vated facilities for the management of ORNL radioactive wastes. A technical support base to handle nonroutine operating problems and to evaluate and monitor the operating performance of the various waste management facilities at ORNL is also provided through this activity.

Activities include the overall management and long-range planning for the improvement of radioactive waste operations, identification and justification of improvement projects, support for conceptual design of line-item and capital projects, preoperational checkout and startup of new facilities prior to turnover to routine operation, and planning for the management of new types of waste that might arise as the result of new R&D or production programs. This task is divided into four subtasks as follows: (1) Management and Planning, (2) Line Item Project, (3) General Plant Projects (GPP), and (4) Engineering Studies and Development.

8.2.1.1 Management and Planning

This subtask supports all activities associated with overall management of the program. An important function is long-range planning, which is performed through the actions of an ad hoc committee. The new committee appointed in October 1981 consisted of 15 members representing several disciplines and drawn from waste management staff and the staff of waste generating programs. This committee and the five-member working group formed to support the committee were divided into seven subcommittees as follows: Gaseous Radioactive Waste, Liquid Radioactive Waste, Solid Low-Level Radioactive Waste, TRU Waste, Standards and Regulations, Decontamination and Decommissioning, and Corrective Actions. These subcommittees met individually to identify and evaluate

problem areas and discuss solutions and reported periodically at meetings of the full committee. After a series of such meetings, each subcommittee had prepared a draft report that described the attributes and shortcomings of the various components of waste management systems and listed recommended improvements to the systems. The working group then used these subcommittee reports to prepare a summary report that presents a consolidated and prioritized set of recommendations for practices or projects to improve the systems. This report also presents estimates of cost to implement the recommendations. The draft summary report was completed and submitted for review and concurrence by management. After revision to accommodate comments and corrections, the report will be issued in FY 1983 as Phase I of the long-range planning activity, to support and guide the preparation of waste management plans that are updated annually. The several subcommittee reports will be issued as a consolidated report that will serve as a source and reference document for planning activities.

Another important function is the coordination of independent assessments of waste management criteria and practices. During FY 1982, a work package was identified for the development of generic assessment criteria for the evaluation of DOE solid waste disposal operations relative to commercial practices and standards. This work package was submitted for competitive bids and was awarded to Dames and Moore. The scope of work also includes use of the generic criteria to assess waste management practices at ORNL and to develop a program plan for assessing each of the other DOE sites administered by Oak Ridge Operations. Dames and Moore submitted a draft report on generic assessment criteria and guidelines; comments and suggested revisions from our review were

resolved in a conference at the end of September, and the final criteria document is being printed. Dames and Moore is using these criteria to proceed with their assessment of ORNL's solid waste management operations.

8.2.1.2 Line-Item Capital Projects

8.2.1.2.1 New Hydrofracture Facility. Construction of this facility was completed early in FY 1982. The facility was dedicated at a ceremony on October 28, 1982, and preoperational testing and a test injection were completed in March 1982. Figure 8.1 shows the facility as viewed from the north side. Details of operation beginning with the ILW injection in June 1982 are given in Sect. 8.2.2 and information on system modifications is given in Sect. 8.2.1.4.

8.2.1.2.2 Radioactive Waste Facilities Improvements Project(s).

The Radioactive Waste Facilities Improvements Project, ORNL 81T-104, will provide new facilities, equipment replacement, and other improvements to the Waste Management System at the Laboratory. The project will provide replacement and upgrading of off-gas and cell ventilation exhaust ducts and related equipment at the main stack; a new building with state-of-the-art electronics and control computer to modernize the Waste Operations Control Center; and new flow measuring devices and support facilities to upgrade monitoring stations located at White Oak Creek, Melton Branch, and White Oak Dam. In addition, the scope of the project was expanded to include upgrading of the isotope area local cell ventilation and hot off-gas systems.

ORNL-PHOTO 4321-82

Fig. 8.1. New Hydrofracture Facility (North Side View).

Waste Operations Control Center. In response to the requirements for improved monitoring and control of radioactive waste effluents, ORNL's Waste Management System is being upgraded. The present Waste Operations Control Center in Bldg. 3105 is being replaced with a new building located on Hillside Avenue. The new facility (Bldg. 3130), completed in September of 1982, will house the computerized data acquisition system for real-time assessment of waste effluent conditions at ORNL. In May 1982, the CADRE Corporation of Atlanta was awarded the contract for a turnkey data acquisition system (DAS) costing \$611,000. The system includes a host computer, seven remote data concentrator microcomputers, field input interfacing equipment, and systems and applications software. The DAS will collect and analyze over 250 field input signals, trend and display these data, and provide the capability for predicting conditions based on historical profiles. CADRE Corporation has entered into subcontract agreements with other firms for supplying parts of the DAS. All computer equipment and systems software have been purchased from Digital Equipment Corporation, data concentrator housings are currently being built by an Atlanta firm, and front-end field signal interfacing modules are being acquired from Tannelec Corporation. CADRE is in the process of developing the applications and communications software. The projected delivery date for the system is March 1983.

Off-Gas and Cell Ventilation Improvements at Main Stack. Improvements

to the 3039 Stack Off-Gas and Cell Ventilation System are to be accomplished in five packages that include: (1) demolition of the old precipitator, (2) building a temporary bypass system, (3) demolition of the old duct and improvements on the scrubber system, (4) building a new cell ventilation system, and (5) removal of the temporary bypass system. The first package has been completed.

Construction of the temporary bypass system is underway by Rust Engineering. This system will be installed to provide ventilation during the demolition of the old duct and the building of the new 3039 Stack Cell Ventilation System. The temporary system includes two loops with fans A-B and C. The ducts for the temporary system have been fabricated and are on site. Most of the ducts for fan A-B that serves the Isotope and 3025/3026 areas have been erected (Fig. 8.2). The temporary fan unit A-B has been set in place as shown in Fig. 8.3. Temporary fan C is also in place.

Upgrading of the off-gas scrubber system, demolition of the old duct, installation of underground piping, and tie-in connections of systems will also be done by Rust Engineering. Modifications to the scrubber (Bldg. 3092) have been completed. The routing of the new steam and air supply lines across Central Avenue has been completed. Fabrication of the new steam control station was started. Procurement of the scrubber, pumps, valves, and other equipment and material is on schedule.

Fig. 8.2. Overall View of 3039 Stack Area Showing Work in Progress on Temporary Duct System.

ORNL-PHOTO 4323-82

Fig. 8.3. Placement of A-B Fan Unit for Temporary Duct System. The 3039 stack is barely out of view on the right and the ORR (Building 3042) is visible north of the stack area at the extreme left.

The contract for the new 3039 stack cell ventilation duct system has been awarded and the FPPC contractor (R. F. Knox Company) given notice to proceed.

Steam Flow Monitoring. The SFM task includes a new monitoring station No. 3 at White Oak Creek, a new monitoring station No. 4 on Melton Branch, and a new monitoring station No. 5 at White Oak Dam. Construction at all three stations is a mix of CPAF and FPPC contracts. Rust Engineering will do the CPAF portion and Vega Corporation will do the FPPC parts.

The major portion of the construction is complete at station No. 3, as shown in Fig. 8.4. The bridge construction over the weir, the installation of instrumentation, and cutting in the stream channel remain to be completed. At station No. 4, the FPPC contractor has completed excavation of the new weir channel. At station No. 5, the FPPC contractor has completed the concrete bridge supports for the new road and has started installations of forms for the remainder of the channel on the downstream side of the road, as shown in Fig. 8.5.

Isotope Area Cell Ventilation and Off-Gas Improvements. In August of 1982, the scope of this project was expanded to include upgrading of the isotope area cell ventilation and hot off-gas systems. This task will provide a new, above-ground stack exhaust stainless steel cell ventilation service to existing cells in eight buildings in the Isotope Area. Also, a new local exhaust will provide building containment and hood exhaust to seven buildings in the Isotope Area. Existing underground ducts will be sealed and abandoned in place.

Design criteria were completed and submitted for DOE review in September 1982. Detailed design is scheduled to begin early in CY 1983.

ORNL-PHOTO 4211-82

Fig. 8.4. New Monitoring Weir (Station 3), White Oak Creek, From Upper Stream Side.

Fig. 8.5. Construction of Channel, New Monitoring Weir, Station 5 at White Oak Dam.

8.2.1.3 General Plant Projects

This activity funds the preparation of conceptual designs, formal engineering designs for and inspections of construction; safety assessments; quality assurance assessments; quality assurance plans when required; environmental assessments and similar documentations that are required for DOE and/or Laboratory approval for directive and subdirective construction projects related to the management of radioactive wastes at ORNL. The projects are followed through design and construction, and the preoperational testing and startup are carried out under this activity before the facilities are turned over for routine operation.

8.2.1.3.1 Piezometer Wells, SWSA 5. Five clusters of wells have been drilled and cased in SWSA 5 to permit study of three dimensional flow characteristics of groundwater in bed rock. Wells of 100-, 150-, and 200-ft depths are common to all clusters. All but the first cluster (that installed by Jack Terry Drilling Company, Sevierville, TN) have a fourth well to groundwater. The wells have been made available to USGS personnel.

All contractual work associated with the project is complete and the project is closed.

8.2.1.3.2 Mag-Card Entry Gate, SWSA 6. SWSA 6 is enclosed by a security-type chain-link fence; the gate, which is near public highway Tennessee 95, is locked on all off-shift hours. However, the gate is open (unattended) during normal working hours to accommodate many trips

in and out by operating personnel. As it is not feasible to man the entrance, a magnetic-card entrance control system was installed by Rust Engineering. The project is complete. Access is available, via plastic card and/or key, to those groups of the Laboratory requiring admission for official business.

8.2.1.3.3 Monitoring Weir, Northwest Tributary (White Oak Creek). This station is to provide quantitative data on activity migrating from SWSA 3 and the Environmental Sciences Complex, Area 1500, to White Oak Creek. It was constructed at the recommendation of the Environmental Management Coordinator as a part of the overall program to increase monitoring capabilities at the Laboratory. Rust Engineering has completed field work on schedule and within allotted funds. Fig. 8.6 shows weir and instrument house.

8.2.1.3.4 Fencing, SWSA 3. The western half of the existing fence encompassing SWSA 3 was replaced as it was beyond economical repair. Approximately 1100 linear feet of fencing were replaced and the associated land surface contouring restored as required. The project was completed within authorized funding.

8.2.1.3.5 Groundwater Monitoring Wells in SWSA Areas (SWSA 5 - TRU Area). USGS personnel working in the ORNL site have included these water table observation wells in their schedule now that all work under this project is complete. Locations for the wells were selected by the USGS geologist. Depths vary from well to well. All but one are less than 50 ft deep; it was cored and is 100 ft deep.

Fig. 8.6. Monitoring Weir, Northwest Tributary.

8.2.1.3.6 Decontamination Facility Alterations (Electropolisher). By means of this new facility, ORNL has greatly improved its capabilities of decontaminating and returning to use small metallic equipment items, apparatus, and tools. The project required extensive remodeling of the chemical process mixing station area within the Fission Products Development Laboratory (FPDL), Bldg. 3517. Rust Engineering and ORNL personnel worked to remove and/or relocate existing equipment, provide temporary facilities, install new equipment, and remove the temporary equipment once the new equipment was operable. Except for miscellaneous items, the project is complete. These minor deficiencies are being addressed and will be completed in time for the directive to be closed within the milestone completion date, first quarter of FY 1983. Project schedule was extended twice: the first time from October 1981 to October 1982, and then to the end of October 1982. Work stoppages by craft personnel necessitated the first extension; the addition of a women's change room dictated the second extension. The project costs are within those authorized.

8.2.1.3.7 Contaminated Oil Storage Facility. This task provides for a one-story addition to the Waste Storage Tanks, Bldg. 7830, to house a 5000-gal storage tank and transfer pump. The tank provides capabilities for storing low-level contaminated waste oil from ORNL Operations while awaiting disposal by injection with ILW waste in the New Hydrofracture

Facility. In the past, it was the practice to dispose of this oil by absorbing it on vermiculite and burying it in carbon steel drums. This is no longer acceptable under current guidelines and regulations for effluents. Cousins Construction Company, Oak Ridge, TN, was awarded the contract for this \$330,000 project; ground was broken in September 1982, and work is progressing satisfactorily. Field work is estimated to be 12% complete. All major procurement items are on the site.

8.2.1.3.8 Demonstration Module - Concrete Vaults. This demonstration system is based on concrete boxes (27 total) filled with selected solid waste, immobilized in grout, and being stacked in three layers on a concrete pad in SWSA 6. The array will be earth covered to establish a radiation shield and weather barrier. Figure 8.7 shows schematic view of the project. Since one year is allocated for filling the modules with waste as it is generated, the end of the fourth quarter of FY 1984 was established as the project completion date. Southern Cast Stone, Knoxville, TN, is constructing the symmetrical concrete steel reinforced boxes to ORNL engineering performance specifications. Work is 50% complete and progressing satisfactorily.

8.2.1.3.9 Waste Treatment Plant Modifications (Additions to PWTP). The flowsheet of the plant was modified to halt the generation of sludge.

The project consists of the procurement and installation of a fourth fixed-bed ion exchange column and miscellaneous ancillary equipment, and changing resin in all columns. The fourth column will be similar to the three existing ones.

An existing settling pond, located in Burial Ground 5, was constructed about ten years ago to provide for the retention of low-level

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TYPICAL REINFORCED CONCRETE WASTE CONTAINER

Fig. 8.7. Demonstration Module - Concrete Vaults.

radioactive sludge removed from chemical waste streams from various laboratory operations. At that time, it was estimated that the storage capacity of the pond would be adequate for 30 years' use. However, current environmental regulations preclude continued use of the pond for sludge storage. The problem of sludge production will be eliminated by discontinuing the existing precipitation/clarification process, which produces the sludge. The addition of the fourth fixed-bed ion exchange column will increase the capacity of the ion exchange process sufficiently to handle the liquid discharge from the laboratories and treat the waste to environmentally acceptable levels for discharge to White Oak Creek. Rust Engineering Company is the contractor for this project. Much field work has been satisfactorily completed as scheduled; however, considerable difficulties have been encountered in the fabrication, by a vendor under contract to Rust, of the ion exchange column that is to expand the capabilities of this plant. The column as initially built not only failed the corrosion test but was found to have been fabricated partly of Type 304 stainless steel in lieu of Type 304L. It is anticipated that the column will pass inspection in early November, at which time all corrections are scheduled to be complete. The project stands 38% complete. The milestone completion date of fourth quarter FY 1983 should be met, and funds expended will be within the \$300,000 authorized for the project. Figure 8.8 shows three columns in the southwest room of the building. The fourth column will be added a line.

8.2.1.3.10 Upgrading Building 7824 (TRU Waste Assay Demonstration). The ORNL TRU Waste Drum Assay Facility (TRU-WDAF) is located in an existing structure, Bldg. 7824, which was upgraded to meet program needs.

Fig. 8.8. Process Waste Treatment Plant.

The purpose of this facility is to field test, calibrate, and evaluate a detector system for TRU waste materials which was developed by Los Alamos National Laboratory (LANL). The detector system assays TRU wastes without breaching the containment provided by the steel storage drums. The work of modifying and upgrading Bldg. 7824 is complete and the directive closed. Figure 8.9 shows the entrance to the facility. A neutron interrogation drum assay system (Fig. 8.10) and gamma drum assay system (Fig. 8.11) with associated instrumentation have been installed, calibrated, and checked out. Additional information on the equipment performance can be found under Sect. 10.2.2 of this report.

8.2.1.3.11 New Hydrofracture Facility (NHF) Site Improvements. This project consists of improvements to the surface area immediately surrounding the New Hydrofracture Facility. Included is surface water run-off control, parking, and asphaltic concrete drives. Inasmuch as this facility is of interest world-wide, these improvements could also be considered necessary for aesthetic as well as for operational reasons. Because of the monthly injection schedule, close coordination is required between operating and contractor personnel. Renfro Construction Company, Knoxville, TN, is the contractor; work is scheduled to begin immediately after the October injection.

8.2.1.3.12 Upgrade Melton Valley Maintenance Building. This project calls for improving the prefabricated steel building acquired from construction; it had served as the contractor's main shop during the construction of the New Hydrofracture Facility. ORNL will utilize the unit as a maintenance building. Improvements include an adequate heating and ventilation system for energy conservation, better lighting,

ORNL-PHOTO 3029-82

Fig. 8.9. Entrance of TRU Waste Drum Assay Facility.

Fig. 8.10. Neutron Interrogation Drum Assay System.

ORNL-PHOTO 1523-82

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Fig. 8.11. Gamma Drum Assay System.

storage area for tools, equipment and work materials, general work area, and a decontamination room for use with equipment from the NHP, mainly pump parts. Design is complete and is being estimated for construction.

8.2.1.3.13 Core Storage Building. A single story prefabricated metal building with 12 ft eave height was erected on a 75- by 85-ft concrete pad east of ORNL's 7000 area by the K. B. Reedy Constructing Corporation, Oak Ridge, TN. The facility is for storing earth cores from the many sites in the vicinity of the Laboratory. Work is 99% complete and final costs will be within those approved. ORNL has taken beneficial occupancy and has arranged for transfer of cores from rented storage facilities. The scheduled completion date, November 30, 1982, will be met. It is anticipated that the official identity of the facility will eventually become Centralized Core Storage Facility - Oak Ridge Reservation.

8.2.1.3.14 Study and Estimates for Future GPP Projects. Engineering studies and estimates have been expense funded for:

1. Corrective Action, SWSA 4,
2. Deep Monitoring Wells (Far-Out Wells),
3. Engineered Groundwater Barrier, SWSA 6 (to be an expense project),
4. Upgrade Valving at the Evaporator Facility,
5. Replace ILW Line Serving Building 3019,
6. Volume Reduction Facility,
7. Initial Decontamination Facility,
8. Upgrading Electrical Substation at TURF,
9. Replace ILW Line Serving Building 3517,
10. Engineered Groundwater Barrier, Trench 7,

11. Replace HFIR ILW Transfer Line,
12. Sparge Waste Storage Tanks,
13. Sparge HFIR Settling Pond, and
14. Control Vegetation and Erosion at Eight Service Ponds.

8.2.1.4 Engineering Studies and Development

8.2.1.4.1 Hydrofracture. Hydrofracture was developed at ORNL for the permanent disposal of locally generated intermediate-level waste (ILW) solutions. In this process, the waste solution is mixed with a blend of cement and other solid additives; the resulting grout is then injected into a nearly impermeable shale formation at a depth of 700- to 1000-ft, well below the level at which circulating groundwater is encountered. During the injection, the grout forms a thin, approximately horizontal, grout sheet up to 1000 ft wide. The grout sets a few hours after completion of the injection and permanently fixes the radioactive waste in the shale formation. A process flow diagram is shown in Fig. 8.12; a picture of the hydrofracture injection facility is shown in Fig. 8.1 (Sect. 8.2.1.2.1).

The hydrofracture facility was designed for the disposal of ILW solution; the disposal of resuspended gunite tank sludge is a comparatively recent expansion in the scope of this process. Much development work has indicated that the operating procedure for sludge disposal would be very much the same as for the disposal of ILW solution; the differences in physical properties between a slurry and a solution could be compensated for by using a different solids mix (one with no attapulgite) and a different mix ratio.

ORNL-DWG 81-18667

Fig. 8.12. Process Flow Diagram of an Injection Which Requires About 10 h to Complete.

Preinjection Activities. The Safety Analysis Report for the New Hydrofracture Facility was originally subcontracted. The final version received from the subcontractor was inadequate and had to be extensively revised. The revised report (ORNL/ENG/INF-81/2) was issued with approvals in May 1982.

In conjunction with the Safety Analysis Report, the Operating Safety Requirements were written and approved. In addition, operating procedures and quality assurance documents were written and approved.

Analysis of previous hydrofracture injections indicated that the blending of the set retarder with the other components of the dry mix materials was inadequate. Such incomplete blending could result in the properties of the blended mix varying appreciably and unpredictably. This hypothesis was checked during the cold test injection (Sect. 8.2.2.1) in March and confirmed. Procedures were therefore altered so that future retarder additions would be made to the waste solution instead of the dry solids.

All observation wells were logged prior to the first injection of waste at the New Hydrofracture Facility and subsequent to each injection at this facility. Observation wells at the shale fracturing facility (retired from service in 1980) were pressure tested and, depending on the response to the pressure test, capped or plugged.

Injections. The first operational use of the new facility (in June 1982) was for the disposal of the ILW solution that had been accumulating since the last injection at the original facility (May 1979). In preparation for the injection, 525 tons of dry solids were blended and stored.

The waste solution was sampled, and the appropriate mix ratio was determined. During the injection, a total of about 160,000 gal of waste solution was mixed with the solids blend and injected during a 2-day operation. This waste solution contained about 15,000 Ci of ^{137}Cs , 140 Ci of ^{90}Sr , and 2.6 Ci of long-lived TRU isotopes. The flow of solids was smooth and uniform throughout most of the injection; consequently, excellent control of mix ratio was maintained. The measured radiation levels outside the process cells were equal to or less than background; the levels inside the cells, determined after washup had been completed, were also generally low. Logs of the observation wells indicate that the major movement of the grout sheet was to the north and west. Logs of the observation wells at the old fracturing facility indicated that no grout sheets traveled that far.

A few difficulties were encountered. The injection pump rattled and bumped at high flow rates, presumably because of incomplete deaeration of the grout. In addition, the mixer hopper occasionally plugged and had to be cleared.

The first operational sludge injection in the new facility was made in August 1982. In this injection, 180,000 gal of resuspended sludge was mixed with a dry solids blend and injected. The injected waste contained about 37,000 Ci of ^{90}Sr , 6,800 Ci of ^{137}Cs , 800 Ci of ^{244}Cm , and 93 Ci of long-lived TRU isotopes. The injection was successfully completed, but considerable operating difficulties were experienced.

The major problem was caused by an irregularity of solids flow in the mixer hopper. (This irregularity had been observed during the previous ILW injection but had caused few problems.) During the injection,

the solids would periodically bridge, or partially bridge, in the hopper. Such situations would require the injection to be halted, the mix tub and the piping manifold to be washed, and the hopper to be cleared. Each interruption lasted several hours. A secondary problem arose from the tendency of grouts containing bentonite to gel (form a jelly-like mass) unless kept in constant motion. As a result, infrequently used lines tended to plug unless washed. In addition, gelled grout tended to accumulate in stagnant areas of the mix tub. These accumulations reduced the effective tub volume and led to control problems; they were also quite difficult to wash from the tub during shutdowns.

Several equipment modifications were made prior to the second sludge injection (made in September 1982). These include the following: (1) the screen was removed from the mix tub and the tub was cleaned, (2) the mix hopper was cleaned and a conical flow disrupter was installed in the mix hopper to improve solids flow, (3) the injection pump was repacked, and (4) a new mix jet nozzle assembly was installed (the old nozzle was found to be eroded).

The initial injection pressure for this injection was unexpectedly high (8,500 psi), but this pressure decreased below 4,000 psi within about an hour. About 120,000 gal of waste slurry was injected; this slurry contained about 46,000 Ci of ^{90}Sr , 2,600 Ci of ^{137}Cs , and 64 Ci of long-lived TRU isotopes. No ^{244}Cm was reported. The injection ran exceedingly well. The flow of solids was smooth and even throughout the injection and the control of the mix ratio was, in consequence, very good. No bridging of the dry solids in the mix hopper was observed and

the tendency of the grout to gel in the mix tube was much less evident. The only difficulties that were noted were a tendency of the injection pump to knock at high flow rates and the unexpectedly high initial injection pressures.

Other Technical Support. The state-of-the-art report on hydrofracture that was written for IAEA was completed and forwarded to IAEA for issuance as a document in their safety series. This report has been issued as an ORNL/CF document for internal use. Slight modifications were made to the document sent to IAEA, and publication is now pending.

8.2.1.4.2 Solid Waste Storage Area No. 7 (SWSA 7) Development.

A draft long-range plan for improvement of facilities and practices for radioactive waste management has been prepared, as mentioned in Sect. 8.2.1.1. The ad hoc committee that prepared the report concluded that the useful life of SWSA 6, the present disposal area for solid low-level waste, is six to ten years and that a new area must be developed within this period. The committee recommended two courses of action, as follows: (1) develop SWSA 7 to receive high-activity, low-level solid waste by 1988-1990 and (2) promote development of a central radioactive waste disposal facility for low-activity waste. The development of SWSA 7 is discussed in this section and in Sect. 8.2.5.3, and the design and development support for a central facility is presented in Sect. 8.2.8.

Siting of SWSA 7 within the White Oak Creek watershed has several advantages, as follows: (1) it contains all waste burial facilities presently utilized with reasonable success by ORNL, (2) it contains an adequate water quality monitoring system for basin discharge, (3) portions

of the basin are underlain by formations which are viable as a burial medium, (4) appreciable hydrologic data are available, and (5) the watershed is near to waste sources.

Of the seven rock units that form the bedrock on the Oak Ridge reservation, two have characteristics which make them the most likely choices as a burial medium for LLW. These rock units are the Conasauga and Knox Groups, both of which occur within the White Oak Creek basin. The Conasauga appears preferable because of its performance under the current waste disposal program on the reservation. The Knox residuum, while apparently successful for sanitary landfills in East Tennessee, has not been sufficiently investigated to evaluate its potential as a burial medium for LLW at ORNL.

Four moderately-sized (>20 acres) tracts within the 1375-acre White Oak Creek drainage basin are underlain by the Conasauga and have tentatively been determined to warrant further site characterization studies. Two of these tracts, when considered together, are the leading candidates for these future studies. Site characterization studies are proceeding on these two favored tracts. These tracts already have installed monitoring wells and lie along strikes with the same rock units in which existing disposal facilities are located.

8.2.1.4.3 Volume Reduction Cost Study. Costs for solid low-level radioactive waste (LLW) treatment steps and disposal operations were developed for the quantity and type of LLW generated at ORNL. Publication costs are reported for various combinations of waste treatment process steps, packaging, and disposal operations based around an existing ORNL facility. Both operating and capital cost components were

developed. The capital cost estimates included a capital recovery factory based on capital recovery in 20 years, 10% annual interest, and levelized payments.

The overall costs for waste treatment and disposal varied from \$40 to \$1740 per ft³ of initial solid waste volume. The lower costs were for Type I (<200 mR) rubble waste with minimum processing and disposal operations and the higher for Type II (>200 mR) combustible waste with incineration followed by ash disposal in a concrete box in a concrete vault. The major cost factor in almost all cases considered was capital costs (which accounted for >90% of overall costs in the highest cost option). All costs are very sensitive to the ORNL throughputs of the various type wastes.

These estimates may be used to help identify the more economical options available for future volume reduction efforts at ORNL. They should be useful elsewhere in relative, rather than absolute, terms.

8.2.1.4.4 Study and Estimates. Another conclusion of the draft long-range plan mentioned above was that the intermediate-level liquid waste (ILW) collection system should be doubly contained. A series of Study and Estimates (S&E) have been started to determine how this recommendation is to be met. These S&Es will be used to obtain the necessary funds and establish priorities. The first four S&Es nearing completion cover:

1. Improved operability for the central receiving system in the Bldg. 2531 (ILW Evaporator) area to include direct access to tanks C1 and C2,

2. Double-contained piping from the vicinity of Bldg. 3019 to the central receiving system which is already doubly contained,
3. Similar piping from close to Bldg. 3517, and
4. Replacing the segment of HFIR-ILW line not presently doubly contained.

8.2.2 Liquid and Gaseous Waste System Operations (E. M. King, L. C. Lasher, R. I. Smitherman)

Facilities are operated for the treatment and disposal of radioactive liquids and gaseous wastes generated at ORNL. There are separate systems for the collection and treatment of low-level (process) liquid waste and intermediate-level liquid waste (ILW). The low-level waste is treated at the Process Waste Treatment Plant (PWTP) by an ion exchange process and the cleaned effluent is released. The intermediate-level waste is concentrated by a factor of 20 to 30 in an evaporator. When a sufficient quantity (~100,000 gal) of concentrated waste is collected it is disposed of at the New Hydrofracture Facility (NHF). At the NHF, the concentrated wastes are mixed with cement and other solids to form a grout and are injected under pressure into a shale formation at a depth of about 1,000 ft, where the grout solidifies, for final disposal.

The gaseous waste disposal system collects radioactive gases (cell ventilation, vessel off-gas) from various areas of the Laboratory and discharges them to the atmosphere through tall stacks. The gases are filtered through high-efficiency filters or scrubbed before they are discharged. Continuous operation of the facilities for the treatment and disposal of the radioactive liquids and gases is essential to prevent releases to the environment from currently operating facilities and from old facilities that were used in earlier discontinued AEC and ERDA programs.

The Eastern Filter Test Station, located at Oak Ridge Gaseous Diffusion Plant (ORGDP), makes quality assurance inspections and tests HEPA filters for all DOE installations east of the Mississippi River and, on a cost recovery basis, for other installations. Also, other DOE-requested or -authorized services such as research, development, and laboratory support are provided in the areas of air filtration.

8.2.2.1 Liquid Waste System Operations

Approximately $452 \times 10^3 \text{ m}^3$ slightly contaminated water were processed through the process (low-level) waste system. The Process Waste Treatment Plant (PWTP) continued operation using the Dowex HCR-WZ (50W-X8) resin that was substituted near the end of FY 1981 for the Duolite resin previously used; this eliminated the head end precipitation step and the subsequent sludge handling procedures. Design was completed and piping changes were made in preparation for the fourth resin column in the PWTP; it is anticipated that the column will be installed and operations will begin in FY 1983.

The intermediate-level liquid waste (ILW) system processed about 5000 m^3 of liquid waste through the evaporator facilities. Operations were routine and without incident.

Preoperational testing and calibration of the instrumentation for the New Hydrofracture Facility (NHF) began near the end of FY 1981. A dedication ceremony of the NHF was held on October 28, 1981; the keynote address was given by Dr. G. K. Oertel. Preoperational checks of the instrumentation and individual pieces of equipment continued; and in early February, the solids handling and liquid systems were checked out up to (but not including) the high-pressure pump using about 80,000 lb of mixed solids (cement, flyash, attapulgate and clay). In March, a cold injection using about 200,000 lb of solids and process water was successfully made; the test was for checkedout/proof of the overall system and training of the operating personnel (including Halliburton) in the new

facility. As a result of the tests some minor equipment and operating procedure changes were made prior to the first hot injection. On June 16-17, an injection of about 160,000 gal of concentrated ILQ waste and 1,000,000 lb of mixed solids was successfully made. During the week of August 9-15 the first injection of resuspended gunite tank sludge was made; this included about 184,000 gal of suspended sludge waste and about 1,050,000 lb of mixed solids. And the week of September 20-24, the second gunite tank sludge waste injection was made -- about 120,000 gal of sludge waste and 700,000 lb of mixed solids. Minor equipment and procedure changes made between injections are described in section 8.2.1.4.1.

The gunite tank sludge removal facility (GTSR) was certified for operation in early July and sluicing operations began on July 12. Operations have continued since, and as noted above, two injections of resuspended sludge have been made at the NHF. The report for the GTSR facility is given in section 8.2.4 of this report.

8.2.2.2 Gaseous Waste System Operations

Gaseous waste operations were normal for the year. The average flow through the Building 3039 stack was $63.7 \text{ m}^3/\text{s}$; this system provides both process and building ventilation throughout most of the Laboratory. Gaseous waste operations at other stacks at the Laboratory were normal.

Construction work continued throughout the year on upgrading of the ductwork and air handling equipment at the Building 3039 stack. In February, the old electrical precipitator in the vessel off-gas system was removed and buried. At year's end, installation of the duct for the temporary exhaust system was underway; this system will provide ventilation to the facilities during installation of the new permanent system.

8.2.2.3 Eastern Filter Test Station (K-25)

A total of 3,656 high-efficiency particulate (HEPA) filters were inspected and tested. Of these 211 (5.8%) failed to meet specification requirements. One hundred fifty-four of the nonconforming filters contained only minor deviations, not significantly affecting their performance, and these were accepted by the purchasers on deviation approvals. A total of 3,850 high-efficiency face mask canisters and/or cartridges were inspected and tested and only twelve (0.3%) were rejected.

The Filter Test Facilities Standards Committee met on October 19-20, 1981, in Golden, Colorado; on February 1-4, 1982, in Hanford, Washington; and on May 24-27, 1982, in Oak Ridge, TN. At the last meeting, drafts of NE Standards F3-42, Operating Policy of DOE Filter Test Facilities, and F3-43, Quality Assurance Inspection and Test of HEPA Filters, were finalized and submitted for Government/Industry review.

8.2.3 Solid Waste Storage Management (J. Bolinsky, T. Grizzard, E. M. King)

Solid radioactive waste generated at ORNL is disposed of by shallow land burial (beta-gamma waste) or retrievable storage (TRU waste) at ORNL; also, very small amounts of off-site waste, as approved by DOE on a case-by-case basis, are disposed of. Disposal is in accordance with approved practices and criteria.

All wastes are segregated and packaged at the point of origin by the generators. Transuranium contaminated (TRU) waste is stored retrievably in stainless steel drums in a pit-type concrete block and reinforced concrete building, in concrete casks in a cave-type building, or (for high beta-gamma TRU waste) in stainless steel capsules in wells with a shielding-plug closure. The TRU waste is scheduled for eventual transfer to a Federal repository. Beta-gamma (non-TRU) contaminated waste with radiation levels above 200 mr/hr at the surface of the unshielded container is disposed of nonretrievably in unlined auger holes. Generally, beta-gamma contaminated waste with radiation levels ≤ 200 mr/hr is segregated by the generator into compactible and noncompactible parts with the noncompactible waste being buried as is, and the compactible portion being compacted into bales (at the burial ground) prior to burial. In addition, in order to conserve trench space, low-hazard (suspect) waste is buried in landfill in the burial grounds.

8.2.3.1 Summary of Waste Handled

The total volume of radioactive waste handled during FY 1982 was 1,941 m³. A breakdown of the disposition of this waste is given in Table 8.1.

Table 8.1. Summary of Waste Handled at ORNL Solid Waste Storage Areas for Fiscal Year 1982 (m³)

TRU waste received and stored retrievably	54
Beta-gamma waste received	1,887 ^a
Beta-gamma suspect buried in landfill	238
Beta-gamma noncompactible received and buried	1,370
Beta-gamma compactible received	279
Beta-gamma compactible buried	34
Beta-gamma (compactible and noncompactible) buried	1,404
Total waste handled (beta-gamma and TRU)	1,941

^a About 577 m³ of this were from facility alterations.

8.2.3.2 Highlights of Operations

Efforts to improve appearances at the burial grounds are continuing. All SWSAs were fertilized and low spots were filled and seeded for grass.

During the year, Building 7824 was partitioned and sixty percent of the floor space was renovated and modified to accommodate the TRU Waste Assay Station. The remaining floor space in the building continued to be used for SWSA storage.

The new Case backhoe and a JCB endloader have been added to the heavy equipment inventory, replacing old and worn-out machinery. Also, a new Dodge crew-cab pickup replaced the old Ford pickup.

There was one off-site generated waste shipment received at ORNL during fiscal year 1982 for retrievable storage at SWSA 5 -- a 0.03 m³ (1 ft³) container containing a ²³⁸Pu source.

TRU waste storage included ninety-five 0.21 m³ (55 gal) and three 0.11 m³ (30 gal) drums in Building 7834; ten concrete casks and one 0.8 m³ (30 ft³) metal box were stored in Building 7855.

Current activities by ESD researchers involving assistance by Operations Division personnel include the Experimental Trench Facility (ETF - nine trenches filled with compacted waste for evaluating the effectiveness of liners and grouts in retarding radionuclide movement) and the experimental suspect waste landfill trench study (three trenches filled with suspect waste and a grout injected in the trenches to fill voids and to retard radionuclide movement). Operations Division personnel cut, clear, and maintain access roads for various ESD activities.

8.2.3.3 Volume Reduction Operations

The actions begun and intensified, beginning in early FY 1978, to reduce the volume of solid radioactive waste for disposal have continued. The most effective appear to be: (1) the awareness program for reducing waste volume, (2) better separation of contaminated from noncontaminated waste at the point of origin and locking of the radioactive waste collection dumpsters, (3) separation of the low-radiation-level waste into a compactible (with compaction at the burial ground prior to burial and a non compactible part, and (4) our landfill operation of low-hazard contaminated waste. These actions will continue along with others including charging the generator for disposal of currently generated waste, emphasis

on control of materials entering contamination zones, and better packaging to reduce voids and facilitate handling.

Team audits were made of all facilities generating solid radioactive wastes to review how well the volume reduction program is working and to discuss associated problems; the teams are made up of representatives from the burial ground operating group, Health Physics, and the group operating the facility being inspected. We are receiving excellent cooperation.

We received sizeable amounts of waste this year from construction, renovation, and cleanup jobs; emphasis has been placed on contacts with supervision of these jobs to encourage reduction of the amount of radioactive waste to as low as practical.

The compactor (for compaction of soft waste with a radiation level of ≤ 200 mr/hr of the unshielded packages) was used to compact about 279 m^3 of waste into 81 bales with a volume of 34.5 m^3 for a reduction ratio of about 8.1.

Since October 1977, users of the burial ground have been charged for costs of current disposal operations. Charges include the cost of collection, transportation, compaction, and disposal (burial or retrievable storage) of the waste; and, beginning in FY 1981, costs of environmental monitoring were added to the charges.

The costs to users experienced to date are

FY 1978	\$272,621
FY 1979	287,283
FY 1980	311,156
FY 1981	513,301
FY 1982	521,204

Note: All costs to date include an overhead rate of about 31%.

Table 8.2 is a summary of the waste handled from FY 1978 through FY 1982. Our volume reduction efforts were intensified beginning in early FY 1978.

Table 8.2 Radioactive Waste Handled at ORNL Burial Ground, FY 1978 through FY 1982 (m³)

<u>Period</u>		<u>General Beta-Gamma Radioactive Waste</u>					<u>TRU Waste</u>
<u>Fiscal Year</u>	<u>Received</u>	<u>Buried in Trenches and Auger Holes</u>	<u>Landfill</u>	<u>Compactible Waste</u>		<u>Savings Due to Landfill and Compaction</u>	<u>Stored</u>
				<u>As Received</u>	<u>After Compaction</u>		
1978	2375 ^a	2255	41	88	10	119	65
1979	1955 ^b	1517	117	361	39	439	75
1980	2409 ^c	1866	230	353	39	544	55
1981	2003 ^d	1610	113	318	38	393	62
1982	1886 ^e	1404	238	278	34	482	55

^aOf this, 282 m³ are construction waste.

^bNot including 247 m³ of waste from cleanup of Burial Ground 3; includes 356 m³ construction waste.

^cIncludes 569 m³ of construction waste.

^dIncludes 577 m³ of construction waste.

^eIncludes 594 m³ of construction waste.

8.2.4 Gunite Tank Sludge Removal Project (L. D. Bates, H. D. MacNary, T. N. Tiegs, H. O. Weeren)

The gunite tank sludge removal project (GTSR) provides equipment and facilities for the removal of approximately 350,000 gal of radioactive sludge from six 15 m diam gunite tanks which were constructed and put into service as a part of the ORNL liquid waste system about 37 years ago.

The necessary equipment illustrated in Fig. 8.13 (Dwg. 82-10509) is sequentially installed in each of the tanks as they are cleaned. The sludge removal is accomplished by a circulatory process utilizing tank W-10 as a mixing tank. A bentonite suspension is initially prepared in tank W-10 which is then pumped through a jet nozzle in the tank being cleaned. The sluicing action of the jet nozzle discharge entrains sludge in the bentonite suspension producing a radioactive slurry which is then pumped back to the mixing tank W-10 after passing through a particle size reduction grinder. Circulation continued until the sludge concentration in the slurry has reached the desired level at which point the sluicing is stopped and the slurry contents of tank W-10 are pumped to the New Hydrofracture Facility for disposal. The process is then repeated until all sludge has been removed.

Facility construction continued from FY 1981. The bentonite mixing system and the work and equipment support platforms over tanks W-10 and W-5 were completed. Necessary 24-in. diam holes in the tops of tanks W-10 and W-5 were core drilled. Figure 8.14 illustrates the removal of a core from tank W-10. The core drill is supported by the work platform over the tank. The pump leg, density probe and mixing jet in tank W-10, the TV camera, sluicer, and adjustable pump leg in tank W-5 were installed. The

Fig. 8.13 Gunite Tank Sludge Removal Facility Equipment Installation.

Fig. 8.14 Removal of Access Hole Core from Tank W-10.

slurry pumps (J-02, J-03) and the grinder were installed and the equipment containment enclosures were completed. Remote controls for the pumps, grinder, TV monitor, and the necessary monitoring instrumentation were installed in the control room (trailer). All site construction activities were completed in December 1981. A critical path study of the project schedule and the interfacing injection operations at the New Hydrofracture Facility resulted in the decision to construct a second moveable platform, which was procured and installed over tank W-6 in March 1982.

The necessary support documents including the Quality Assurance Plan, Quality Assurance Assessment, Preoperating Checkout and Calibration Procedures, Operating Procedures, Final Safety Analysis Report (FSAR), and Operating Safety Requirements (OSR) were prepared, reviewed by appropriate groups, approved, and issued. In addition to ORNL approval, the FSAR and OSR were reviewed and approved by DOE-ORO.

Instrument checkout and preoperational testing began in February and extended through June. Minor modifications in equipment and operating procedure were made as a result of the testing program. The Office of Operational Safety and the Radioactive Operations Committee conducted a preoperational review in late June followed by ORO review in early July. Both groups promptly indicated approval to initiate operations; sluicing began on July 12.

Sluicing and periodic transfer of the suspended sludge to the Hydrofracture Facility continued to the end of the year. Early sluicings produced waste batches with undesirably low solids content. In September, a redesigned screen was installed on the suction leg in tank W-5, and a

hydraulic pump was installed on the movable section of the leg so the screen could occasionally be raised for washing with the sluicer. This change has resulted in a favorable increase in slurry solids concentration.

As noted in section 8.2.2, two injections of suspended sludge waste have been made: approximately 184,000 gal the week of August 9 through August 15, and approximately 120,000 gal the week of September 20-24. Operations are currently continuing in tank W-5; and tank W-6 is being readied to receive the sluicing equipment when tank W-5 is empty.

8.2.5 Site Specific Low-Level Waste Studies. (D. D. Huff, J. Switek, C. R. Olsen, C. S. Haase, F. G. Taylor, P. D. Parr)

The primary purpose of the Site Specific Low-Level Waste Studies task is to provide environmental research and development support to solve low-level radioactive waste management problems at ORNL. The major objectives of the work are to:

1. Identify environmental factors that affect waste disposal operations at ORNL,
2. Determine the potential impact of waste disposal operations on environmental safety and regulatory compliance, and
3. Determine and recommend procedures to ensure that environmental safety and regulatory goals are met.

The following subtask descriptions present summaries of activities directed toward meeting the objectives described above.

8.2.5.1 SWSA 4 Studies

The primary purpose for SWSA 4 investigations is to supply information needed to satisfy the objectives of upgrading and achieving long-term stabilization of the burial ground. Since SWSA 4 is the most important contributor of ^{90}Sr to White Oak Creek (8.1), it has been the focus of detailed studies of the distribution of ^{90}Sr as well as mechanisms for the hydrologic transport of ^{90}Sr from disposal trenches (8.2). These studies have resulted in plans for a surface water diversion to capture uncontaminated runoff from the up-slope portions of the SWSA 4 watershed and divert it to other basins. Construction

is expected to be completed in 1984. In the meantime, more detailed characterization of the nature and extent of radionuclide contamination susceptible to transport by groundwater runoff from within SWSA 4 has proceeded. A 30 x 30 m grid system has been extended into the region of most active ^{90}Sr transport and is being used to accurately determine the spatial distribution of radionuclide distribution there. A surface contamination survey on 5 x 5 m centers has revealed patterns that suggest releases from several individual sources rather than a general broad area source that might be expected as a result of contaminated groundwater discharge over an extended area. Soil samples collected using a percussion-driven coring device have been subjected to gamma-scintillation analysis and show presence of both ^{137}Cs and ^{60}Co . Cesium-137 contamination is present primarily at the ground surface. Cobalt-60 shows highest concentrations at 0.3 to 0.5 m below ground at locations where it is present. The nature of the 3-dimensional distribution pattern strongly suggests that the spread of contamination results from the "bathtub" effect within several individual trenches in low-lying areas. These trenches, which were constructed perpendicular to contour lines, collect water and overflow to the ground surface at the downslope end rather than releasing contaminated water to the surrounding soils for subsequent groundwater transport. It is expected that this shortened pathway will be reduced to a minor role by the diversion of upslope runoff when the diversion construction is completed in 1984. However, evaluation studies are continuing and the need for additional stabilizing measures will only be

determined for certain after the surface runoff diversion project has been completed.

8.2.5.2 SWSA 3 Studies

Stueber et al. (8.3) noted that Solid Waste Storage Area 3 appeared to supply small quantities of ^{90}Sr not only to the Northwest Tributary (NWT) of White Oak Creek but also to Raccoon Creek which lies outside the White Oak Creek watershed. The suspected transport pathway was by way of a solution cavity labyrinth that parallels the geologic strike. In order to further delineate the nature of this solution network and to allow the development and demonstration of corrective measures to be used in shallow land burial grounds sited in carbonate (limestone or dolomite) rocks, further characterization using dye tracer tests, water chemistry analyses, and geologic studies has been pursued.

8.2.5.2.1 Dye Tracer Tests. Dye tracer experiments have confirmed that SWSA 3 is the source of radiostrontium to seeps on Raccoon Creek and the Northwest Tributary (NWT) to White Oak Creek. The relative intensity of dye reaching the NWT and the Raccoon Creek seep (Fig. 8.15) indicates that a spring on the NWT (the D-270 seep) is the major recipient of radionuclides migrating from SWSA 3. This spring should be an excellent monitoring point for assessing the effectiveness of any remedial actions implemented at SWSA 3. Breakthrough of tracer to both Raccoon Creek and NWT reveals that the ground drainage water divide is approximately 100 m further east than previously believed. It now appears that the entire southwestern portion of

Fig. 8.15. Map of the SWSA 3 area, ORNL, showing contaminated seeps and the appropriate location of the Rome-Chickamauga fault clay deposit.

SWSA 3 contributes groundwater to the Raccoon Creek drainage under appropriate conditions. The fact that the concentration of ^{90}Sr discharged to Raccoon Creek (2 Bq/L) represents only 1% of the corresponding concentration in water discharged to NWT (200 Bq/L) may be misleading in that the distance from the injection well (≈ 900 m) to the Raccoon Creek seep is almost double the distance to the NWT seep (500 m) and dilution and dispersion may be considerable. Samples collected at a point on NWT approximately 900 feet from the injection well contained ^{90}Sr in concentrations of the same order of magnitude as those collected at the new Raccoon Creek seep (8.3). On the other hand, it must be noted that stream discharge at Raccoon Creek seep is much less than at Stueber's sampling point on NWT.

8.2.5.2.2 Water Chemistry Analysis. Sampling of SWSA 3 wells and streams has shown that most waters in the vicinity are of a calcium-bicarbonate type as is expected in a limestone aquifer system. However, several wells (including all wells within the SWSA 3 fence) show elevated levels of sodium, potassium, sulfate and chloride. This presumably represents contamination from waste trenches but it could also be due to other activities such as addition of fertilizer to the grass. Well-bore muds will be analyzed to further evaluate this situation. Radionuclide analyses of the SWSA 3 area waters has shown that almost all the wells surrounding the site can contain detectable ^{90}Sr and several contain detectable ^{60}Co and alpha contamination. This indicates that the groundwater flow system within and around the burial ground is quite complex although on a larger (10^2 m) scale

the system appears to be dominated by a small number of solution cavity flow paths.

8.2.5.2.3 Geological Studies. Reconnaissance level geological investigations around SWSA 3 suggest that the southwestern end of the area is underlain by fault gouge-clay that marks the thrust faulted contact between limestones of the Chickamauga Group and the older, overlying Rome Formation (Fig. 8.15). Stratigraphic cross sections prepared from logs of SWSA 3 wells drilled in 1949-50 do not show significant deformation of Chickamauga rocks related to this thrusting event. The contact between the clays of the fault gouge and outcropping limestones on the eastern side of SWSA 3 appears to be a major groundwater flow zone feeding water directly into the middle of the burial ground from a small catchment area on Haw Ridge. The area at the base of this catchment, just outside the fence, is an old landfill with one trench about 100 m long (not shown). The disturbed ground probably increases the amount of water storage in this area and releases water to the burial ground over a prolonged period of time. It may be possible to install a grout curtain or french drain between this fill area and SWSA 3 to direct water away from the buried wastes. Alternatively, a drain could be installed at the top and a bentonite cap placed over the landfill itself. Future studies will involve drilling a well into the area and conducting a dye tracer test to establish the hydrologic connection between this area and NWT. Seismic surveys will be run to determine the position of the clay-limestone contact more accurately.

8.2.5.3 SWSA 7 Siting and Characterization

Within approximately 10 years new disposal facilities for solid low-level radioactive waste will be required at ORNL. SWSA 7 is intended to receive higher activity low-level waste and characterization of a proposed site has begun. Groundwater levels in the proposed SWSA 7 were monitored in 11 wells throughout the year. Seven of the original wells were redrilled to 30 m and four of these were finished with multiple screens to permit study of deeper levels of groundwater flow. In September, three additional 30 m wells with multiple screens were installed. Thus, by September 30, 1982, groundwater monitoring was being conducted with a total of 14 wells, 4 of which are 10 m in depth and 10 of which are 30 m in depth. In addition, a rain gauge has also been installed to measure precipitation at the site.

Making use of the observed minimum depth to groundwater level from calendar year 1981 (measured from land surface), and land surface slope data a preliminary map illustrating the potential usable land areas for conventional shallow land burial practices has been constructed (Fig. 8.16). The cross-hatched regions in Figure 8.16 represent potentially usable area where the depth of groundwater is greater than five meters and the slope is 25% or less. The size of the region is approximately 10 hectare (~ 24 acres). This value is very dependent on the minimum acceptable depth to groundwater value assumed in construction of the map. If a value of two or three meters for this parameter is acceptable then the usable area would

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Fig. 8.16 Map showing the general area encompassed by proposed SWSA 7. Regions enclosed within the solid lines in the center of the figure are potentially usable portions that have a depth to groundwater greater than five meters. Within these regions the cross-hatched portions are areas where the slope of the land surface is less than 25%.

increase significantly, perhaps by a factor of two or three. The appropriate value for this parameter and the value of the maximum acceptable surface slope will be determined by the disposal technology to be employed at SWSA 7. The extent of the potentially usable region illustrated in Figure 8.16, therefore, must be considered as tentative at this stage of engineering and design for SWSA 7.

To provide measurement of surface-water discharge from SWSA 7, a critical-flow flume with complex cross section has been designed, fabricated and installed on the primary stream that drains the proposed disposal area. The flume was designed to provide accurate measurement of discharge over a wide range in flow rate and at the same time cause minimal ponding of water upstream. The design parameters were finalized using an interactive computer program (8.4) together with field-determined estimator of downstream channel capacity. Since the flume represents an experiment in the performance of low head-loss measuring structures, it was fabricated from plywood to minimize costs. It has been installed on a concrete foundation that should allow for changes in design, should they be found necessary.

Early indications are that the flume performs well enough to measure flow within $\pm 3\%$ at very low flow rates. Evaluation of the design will continue and be the basis for more permanent structures in future years.

8.2.5.4 ILW Pits and Trenches

From 1951 to 1966 seepage pits and trenches were used for the disposal of intermediate-level liquid wastes (ILW) at Oak Ridge National Laboratory (ORNL). These waste pits and trenches were excavated in weathered Conasauga Group and backfilled with crushed limestone. Waste liquids (adjusted to a pH of ~ 12 with NaOH) were allowed to percolate through the weathered rock (pH ~ 5) and radionuclide migration was retarded by reactions with the fill and soil. Over one million curies of fission products ($\sim 75\%$ ^{137}Cs and $\sim 25\%$ ^{90}Sr) with some activation products, actinides and transuranics were disposed in this manner prior to the implementation of deep-well hydrofracture.

Groundwater samples have been collected on a monthly basis from ~ 35 wells in the vicinity of these formerly used ILW pits and trenches. These samples have been routinely analyzed for pH, ^3H , ^{90}Sr , ^{137}Cs and ^{60}Co . Selected samples have also been analyzed for other radionuclides (including ^{99}Tc , ^{233}U and transuranics) and water chemical characteristics (ECO_2 , Eh, PO_4 , Fe, Mn, etc.). Data concerning the chemical properties of the water have been used to help identify the geochemical factors and complexes which may influence nuclide solubility. Once in solution, nuclide migration from these waste disposal sites is controlled primarily by local hydrological flow patterns which are governed to a great extent by local geological features, such as folds, faults, and zones of greater weathering.

Concentrations of ^3H , ^{60}Co , ^{233}U , ^{238}Pu and ^{99}Tc in the groundwaters around the ILW pits and trenches may range as high as 12,000 Bq/L, 1700 Bq/L, 41 Bq/L, 0.06 Bq/L and 3200 Bq/L, respectively. The distribution of ^3H has been used to delineate contaminated flow paths. Complexation of ^{60}Co with a soluble chelating agent (EDTA) has been the reported cause for its high mobility. Experiments to positively identify the chemical speciation of mobile ^{60}Co are presently in progress. The mobile ^{233}U species is a low molecular weight anionic complex, probably a carbonate complex, but the possibility of a phosphate complex has not been ruled out. Between 20 and 50% of the dissolved ($< 0.45\ \mu\text{m}$) ^{238}Pu in the groundwaters is in an oxidized (V, VI) state. It is suspected that the ^{238}Pu contamination in these groundwaters actually reflects the movement of its parent ^{242}Cm at the original time of disposal rather than the subsequent migration of ^{238}Pu . The high concentrations of long-lived ^{99}Tc (210,000 y half-life) in the groundwaters probably reflects its mobility as a negatively charged pertechnetate ion, TcO_4^- , which is stable in slightly oxidizing aqueous solutions. At present, levels of ^{90}Sr contamination in these groundwaters are relatively low ($< 1\ \text{Bq/L}$) as a result of retardation by reactions of the alkaline waste with Ca-Mg carbonates in the soil or crushed limestone fill.

A detailed site characterization of the geochemical, hydrological and geological factors governing radionuclide migration is presently being undertaken at ILW Trench 7. This site was chosen because of the occurrence of ^{60}Co and ^{233}U in a surface seep which

appears in a valley ~ 50 meters east of the trench. Groundwater ^{60}Co and ^{233}U concentrations undergo a seasonal variation at this site with the highest activities occurring in the springtime. The seasonal rise in nuclide activity is correlated with a rise in the groundwater level and a concurrent increase in groundwater pH. This suggests that during spring, the water table may rise to saturate the alkaline fill in the trench or contaminated soils near the trench and thus promote nuclide migration.

Two suspected migration pathways of contaminated water from the trench to the nearby seep have been identified on the basis of nuclide concentrations, inflections in the groundwater table, groundwater compositions, field seismic surveys and local geologic structures. Both pathways appear to be associated with fault zones which strike parallel to a line drawn from the trench to the seep area. A series of wells drilled parallel to the trench and across these fault zones (open circles in Figure 8.17) indicated an inflection in the groundwater table, rising sharply to the north of well T7-23. Seismic refraction results were too complex for unequivocal interpretation but definitely indicated structural discontinuities at T7-21 and T7-23. Along this series of wells, a geologic cross section was exposed parallel to the trench by widening the banks of an already existing access road. This exposure delineated a fault zone at T7-21, a large anticlinal fold at T7-23 and another fault zone at T7-25 (Figure 8.18).

Groundwater analyses indicated that the greatest contamination occurred in wells drilled through the fault zones and little or no

Figure 8.17. Areal map illustrating the location of previous wells (closed circles) and new wells (open circles) in the vicinity of ILW Trench 7.

contamination in the groundwater capped by the limestone fold (Table 8.3). Well T7-21, which is not shown in Figure 8.18, is a 15 meter well which penetrates a fault zone ~ 8 meters to the south of well T7-22. This fault zone is delineated by a topographic gully, along which several shallow 9 meter wells (T7-4, T7-9, T7-10 in Figure 8.17) were drilled over a decade ago. These shallow wells did not penetrate deep enough to tap the contaminated flow path as indicated by gamma-ray borehole logs and a comparison of the nuclide data for wells T7-21 and T7-10 in Table 8. Although ^{60}Co and ^{99}Tc concentrations in groundwaters of T7-21 are high, the alkalinity, pH and ^{233}U concentrations are relatively low suggesting that this pathway is not presently transporting contamination from the trench itself but rather is a relict pathway which was active during past disposal operations.

The high alkalinity and pH of the groundwaters in wells T7-24 and T7-25 along a fault zone at the north end of the trench (Figures 8.17, 8.18) in addition to the high ^{60}Co , ^{99}Tc and ^{233}U concentrations (Table 8.3) indicate that this pathway is presently transporting contamination from the north end of ILW Trench 7 to the seep area in response to the spring rise in the groundwater level. A new series of wells and soil borings are presently being drilled at the north end of the trench to evaluate the above mechanism and pathway for nuclide mobilization and transport, and the effectiveness of a french drain, groundwater interceptor as a remedial action plan to retard nuclide migration along this pathway.

Table 8.3. Contamination Levels in Wells Near ILW Trench 7¹

Well Number	pH	³ H (Bq/L)	⁹⁰ Sr (Bq/L)	⁹⁹ Tc (Bq/L)	⁶⁰ Co (Bq/L)	²³³ U (Bq/L)
T7-10	6.5	40±100	0.4±0.2	--	89±3	0.2±0.1
T7-21	6.9	29000±1000	< 0.1	3200±100	2680±67	1.6±0.2
T7-22	7.3	10000±1000	< 0.1	--	1090±41	0.3±0.1
T7-23	7.1	10000±1000	0.9±0.3	--	< 4	< 0.1
T7-24	7.3	6700±1000	< 0.0	400±10	2180±11	2.3±0.3
T7-25	8.9	5600±1000	1.6±0.4	--	1820±30	15.0±2.0

¹The samples were collected on April 23, 1982. The ⁹⁹Tc data are for samples collected on July 23, 1982.

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Figure 8.18. Cross-sectional photo and map illustrating the geological structures which strike between ILW Trench 7 and a nearby contaminated seep. Several wells (for which radionuclide data are presented in Table 1) are located in relation to the geologic structures.

8.2.5.5 Long-Term Vegetation Management

The use of vegetative covers other than grass or change in maintenance of the current grass cover may be advantageous for both short-term and long-term management of former solid low-level nuclear waste disposal sites. The "ideal vegetation cover" would require low or no maintenance, have a high water demand, a shallow root profile, cause negligible radionuclide uptake, and be capable of successful competition with invading successional species.

Plant species with potential for use as ground cover on the low-level waste disposal areas were identified through a literature review. Vegetational types evaluated included successional grass stages, successional thicket stages, shallow-rooted lawn grass species, drought-tolerant lawn grass species, prairie grass species, herbaceous horticultural species, forbs and weeds, low shrub species, tall shrub species and trees. No one particular species or vegetational type emerged as fulfilling all requirements for the "ideal" cover.

Based on the literature review, natural succession stages may be promising, although the stabilizing of particular stages may require some maintenance. Broomsedge fields, for example, will not require fertilization, have a sufficiently shallow root profile, and are relatively stable if mowed annually. Thicket or vine stages (e.g., honeysuckle, kudzu, sumac) may cause problems with regard to invading deep-rooted species. A grass cover (as presently maintained on the disposal areas) requires greater labor-intensive (mowing, fertilization) care, but can provide the least risk of root mining as long as deep-rooted, drought resistant species are avoided. A dense thicket of ornamental

shrubs (sumac, spiraea, forsythia, jasmine) can be obtained if shrubs are pruned back occasionally. Of the herbaceous ornamental species, yellow root, groundcover, bamboo and lilyturf or a combination of these with periwinkle, bugleweed, and English ivy can be effective. Ornamental shrubs and herbs will require some maintenance also (periodic weeding, pruning, pest control).

Some possibilities may exist for low-maintenance arboreal vegetation. For example, preliminary work with simulation models indicates plantations of certain early successional species may provide suitable vegetation cover while requiring infrequent, labor-intensive manipulation. Species such as red maple or white ash may have proper rooting characteristics, sprout readily when harvested, and can regenerate and maintain, through coppicing (stump sprouting) on a 20-year cycle, leaf areas suitable for relatively high evapotranspiration. Current work with the modeling effort also indicates highly variable growth under different moisture and nutrient regimes.

The long-term objective of the research is to demonstrate that improved vegetative covers are available and to provide a sound basis for recommending the type of cover best suited for a particular disposal area. This will be evaluated locally through a series of demonstration plots and extrapolated regionally through modeling efforts utilizing forest types and variable environmental conditions typical of the Southeast.

8.2.5.6 Melton Valley Geology

Research this past year has focused on the regional stratigraphic character of the Conasauga Group throughout east Tennessee and on the refinement of the general stratigraphic and petrologic understanding of the Conasauga Group at the Oak Ridge locality. The purpose of this subtask is to characterize the sedimentology, stratigraphy, and petrology of the Conasauga Group so that its role as a host formation for the hydraulic fracturing and the shallow land burial operations at ORNL can be better evaluated.

Regional stratigraphic studies were based on an extensive compilation of data from published and unpublished sources. Lithofacies and isopach maps in conjunction with stratigraphic cross sections of the Conasauga Group and its constituent formations in east Tennessee were prepared as both present day and palinspastic bases (8.5).

Examination of the present-day base isopach and lithofacies maps resulted in the identification of spatial and temporal patterns of lithofacies distribution that are generally consistent with those outlined by Rodgers (8.6). Such patterns suggest that the Conasauga Group was deposited on a passive continental margin in a shallow marine shelf setting. Examination of the data plotted on a restored palinspastic base, which removes the effects of depositional thrust faulting, demonstrates that the pre-thrusting isopach and lithofacies patterns are not consistent with the simple shallow marine shelf depositional setting. Instead the data suggest the following features for the regional depositional setting for the Conasauga Group:

(1) depositional settings ranged from dominantly marine to locally peritidal and sedimentation occurred on a northeast-southwest trending continental shelf (300-500 km wide) that contained an intra-shelf basin (100-200 km wide) that closed to the southwest and opened to the northeast along an axis approximately parallel to the trend of the shelf margin, (2) the eastern margin of this intra-shelf basin consists of shaly limestones and dolostones that become dominantly dolostones eastward of the basin margin; (3) intra-shelf basin sediments consist mainly of shales, siltstones and shaly limestones; and (4) the western margin of the basin consists mostly of siltstones, shales and calcareous shales that become dominantly clastic-rich westward of the basin. These conclusions about the Conasauga Group depositional setting and lithofacies distribution patterns for the east Tennessee region agree well with the shelf-intrashelf basin-carbonate ramp shelf margin model proposed for the Nolichucky shale and upper Conasauga Group equivalents in southwest Virginia by Markello and Read (8.7). The results of this study provide information about the southern extent of the intra-shelf basin described by Markello and Read (8.7) and about nature of the Conasauga Group depositional setting southward of the area studied by them. Results of this regional stratigraphic study indicate that the Conasauga Group that occurs at the Oak Ridge locality was deposited on the western edge of the intra-shelf basin but eastward of the western basin margin. The abundance of clastic-rich material and the complexly interfingered nature of clastic-rich and carbonate-rich lithologies within the

Conasauga Group at the Oak Ridge locality are consistent with such a depositional setting.

Local stratigraphic investigations of the Conasauga Group in the White Oak Mountain and Copper Creek thrust blocks to date (8.8) have been based on limited outcrop data and widely scattered diamond drill cores. To provide a complete reference stratigraphic section of the Conasauga Group for the Copper Creek thrust block (Melton Valley) a deep borehole drilling project was begun in August of 1982. The goal of this project is to obtain a continuous rock core of the Conasauga Group and the underlying Rome Formation. The rock core will allow an unambiguous correlation of numerous fragmentary rock cores of Conasauga Group material obtained previously from throughout the DOE Oak Ridge Reservation. It will also provide stratigraphically well-characterized samples for use in geophysical well logging studies planned for FY 1983. As of October 26, 1982, the deep borehole had been drilled to a depth of 730 m (~ 2400 feet); total depth of the borehole will be approximately 900 m (~ 2900 feet).

8.2.6 Hydrofracture Mix and Grout Development Studies (E. W. McDaniel, L. R. Dole, C. A. West, and D. P. Stinton)

During FY 1982 the following four tasks were addressed: (1) grout mix development studies; (2) leach studies; (3) injection assistance; and (4) nuclide migration studies.

8.2.6.1 Task 1: Grout Mix Development Studies

Studies were conducted to evaluate different grinds of attapulgite clay as a suspending agent for ILW grouts.

Test samples of Attagel-350 clay were evaluated as a substitute for the Attapulgite-150 drilling clay presently used in hydrofracture grouts (Table 8.4).

There are no important differences in the rheological parameters of the grouts prepared with these clays. The materials are essentially the same, but they were given different names and directed at different markets.

Another grind, X-8000, was evaluated but produced grouts with an unacceptable phase separation (>20 vol %).

Experiments were performed to determine the effect of inhomogeneously mixing dry solids on grout rheology. The data shows that ± 14 wt % variation in cement or fly ash have little effect on phase separation. Fluid consistency index (K') varied from 1.72×10^{-3} to 2.28×10^{-3} . The flow behavior index (n') ranged from 0.56 to 0.60. Increasing the attapulgite clay by 38 wt % decreased the phase separation from $\sim 17\%$ to $\sim 8\%$ and decreased n' from 0.60 to 0.34. (Note: A small value of n' means that the grout is thixotropic and shear thins.)

Table 8.4. Properties of grouts¹ made from different trade names of Attapulgite clay

Rheological parameters	Trade name ²	
	Attapulgate-150	Attagel-350
Fluid consistency index (K')	1.23×10^{-3}	2.15×10^{-3}
Flow behavior index (n')	.60	.58
Apparent viscosity (mPa·s)	7.5	8.0
Density (g/cm ³)	1.40	1.41
Phase separation (vol %)	11.80	9.5

¹Standard dry solids blend dry sheared for 2 h at 5000 rpm in Waring blender before grout preparation. Grouts were mixed with W-7 simulated waste solution in the ratio of 719 g/L at 2000 rpm.

²Trade names of Engelhard Minerals and Chemical Corporation, Edison, NJ.

A preliminary conclusion to be drawn from this work is that it is necessary to have solids blended as homogeneously as possible to insure acceptable grout rheology.

8.2.6.2 Task 2: Leach Studies

Strontium leach studies of hydrofracture sludge-slurry grouts were completed and reported in ORNL/TM-8198 in March 1982. The results obtained in this study indicate that use of the modified IAEA leach procedure and the Joy-Godbee model gives an apparent diffusion coefficient of $5 \times 10^{-12} \text{ cm}^2/\text{s}$ and a moving boundary coefficient of $1 \times 10^{-7} \text{ s}^{-1}$ for one of the best grout formulations in distilled water (Fig. 8.19).

8.2.6.3 Task 3. Injection Assistance

The major effort in this task is to determine the dry solids blend and amount to be used for a given waste stream. This is performed prior to the injection. Samples of dry solids are taken at specific intervals during an injection and grouts are prepared with a simulated waste as a quality assurance measure. Based on viscosity calculations from data on the sludge-suspensions that were transferred from the south tank farm to the new hydrofracture (NHF) site holding tanks, the amount of dry solids per gallon of sludge to be used was determined using a rheogram that relates flow properties of these sludge-slurry grouts to previous pumpable grouts made with waste solutions.

Grouts were prepared with dry solids taken from the hydrofracture site's weighbelt feeder, at approximately 46,000 Kg intervals during the first injection of gunite tank sludge (Aug. 9-15, 1982). Grouts

Fig. 8.19. Fraction of strontium leached vs time from sample 280005.3, cured 182 d.

were made from these samples as a quality assurance (QA) verification of quality control (QC) procedures, which were followed during the dry solids blending operations. Characteristics of these grouts (Table 8.5) show that the solids in T-8 and T-9 silos were blended uniformly and verify that the site operations conscientiously maintained the dry solids blends within the QC limits during preparation for the SI-1 injection.

8.2.6.4 Task 4. Nuclide Migration Studies

Studies were initiated in FY 1982 to characterize hydrofracture grouts in terms of phase composition and microstructure. Numerous grout samples containing simulated waste solutions or sludges are currently being investigated by optical microscopy, X-ray diffraction, and scanning electron microscopy. Fracture surfaces, polished surfaces, and crushed samples are being examined to identify the phases present. We are currently learning to identify fly ash, calcium hydroxide, and the cementitious phases of tobermorite, ettringite, calcium aluminoferrite hydrate, and tetracalcium aluminate hydrate.

A core-drilled sample of hydrofracture grout injected nearly 18 years ago was examined by optical microscopy, scanning electron microscopy, and X-ray diffraction. The X-ray diffraction results revealed that the grout consisted primarily of calcite (CaCO_3), vaterite (CaCO_3), portlandite [$\text{Ca}(\text{OH})_2$], and Ca_2SiO_4 . Fortunately grout samples prepared in the laboratory produced the same phases as actual grout sheets injected into the shale layer and cured in-situ. The core-drilled sample had no quartz or mullite present indicating that no fly ash had

Table 8.5 Comparison of grout properties made with dry solids¹
taken at different places in T-8 and T-9 silos during
SI-1 injection

Grout ² parameter	T-8			T-9		
	Amount in silo (Kg)			Amount in silo (Kg)		
	136,363	100,000	31,818	106,818	86,363	45,454
Apparent viscosity (cP)	35.5	34.0	35.5	36.0	35.0	34.0
Density (g/cm ²)	1.45	1.44	1.44	1.45	1.44	1.44
Phase separator (Vol %)	3.70	3.75	3.75	3.74	3.75	4.16

¹Dry solids composition 46 wt % type I cement, 46.0 wt % fly ash and 8.0 wt % pottery clay.

²Mixed in the ratio of 719 g of dry solids per liter (5.5 lb/gal) of a 15 wt % sludge that has a viscosity of 9.0 cP.

been added to the grout. Because fly ash was added to all injections since experimental injection six, the core-drilled sample must be from one of the first five experimental injections. Further analytical examination identified the radionuclides as ^{90}Sr , ^{137}Cs , and ^{60}Co . The relative amounts of these radionuclides identified the grout sheet as experimental injection five.

A portion of the core-drilled sample was mounted for metallographic examination (Fig. 8.20) and found to contain numerous fragments of shale trapped in the grout. Examination by scanning electron microscopy could not locate the Sr, Cs, or Co because of the very low concentration. However, a β - γ autoradiograph of the sample clearly showed the location of the radionuclides, and more importantly that the radionuclides had not migrated from the grout into the included shale fragments in the 18 years since injection.

Fig. 8.20. Beta-gamma autograph of cored grout shows no migration of radionuclides.

8.2.7 Waste Volume Reduction, F. N. Case, J. R. DeVore, T. F. Donaldson, K. W. Haff, S. E. North, J. C. Posey, R. W. Schaich, F. J. Schultz, C. D. Scott, G. W. Strandberg, and J. A. Tompkins

8.2.7.1 Dissolution of HFIR Control Plates

This dissolution of old stored HFIR control plates and their ultimate disposal by hydrofracture is desired to eliminate the storage requirement and to provide space for future change-outs of these control plates.

Two different methods for the dissolution of the aluminum metal-europium oxide plates were tested successfully. The first method was the direct dissolution of the aluminum and europium oxide in nitric acid using a mercury catalyst. The second method consisted of two steps: first the aluminum was dissolved in sodium hydroxide solution; then, nitric acid was added to convert the aluminum to the nitrate and to dissolve the europium oxide.

It was shown that the use of iron transfer cans for the control plates is not feasible. The cans containing the plates will be dropped into the dissolution solution. Therefore, both the cans and plates must be soluble. Iron will not dissolve in sodium hydroxide solution. Furthermore, although the iron dissolves rapidly in nitric acid, its presence in solution greatly slows the dissolution of aluminum. Aluminum cans should be used.

The use of hydrazine to suppress the formation of nitrogen oxides in the mercury catalyzed nitric acid reaction was abandoned. On three occasions an unexpected rapid reaction took place between the nitric acid and the hydrazine. These reactions were not explosions; they required several seconds to go to completion. The very rapid evolution of gas and heat boiled most of the liquid out of the beakers in which the tests were being conducted. On other occasions no reaction was observed even during

prolonged boiling of the solutions. Rapid dissolution of the aluminum was achieved and observation indicated that neither NO or NO₂ was produced.

In the best test of the aluminum-nitric acid reaction the initial nitric acid concentration was 5.2 M and the mercury concentration was 0.4 g/liter. There were 46 liters of solution per Kg of aluminum. These values are equivalent to 102 liters of 15.5 M nitric acid and 121 g of mercury per control plate and aluminum transfer can. The equivalent total volume is 303 liters. This is well below the 1779 liter capacity of the tank in which the reaction will be carried out. Further research may allow some reduction in the reagent requirements.

In this test the mercury catalyst was added to the solution at 64°C. A very rapid reaction started which quickly heated the solution to boiling, 108°C. After 9 minutes most of the aluminum had dissolved. The remaining part of the sample was floating supported by the bubbles generated on its surface. Only after 38 minutes more had the last trace of the sample dissolved. The total time was 47 minutes. All of the very fast reaction, both acid and caustic, followed this pattern. An initial period of very fast reaction was followed by a longer period of slower reaction with a floating aluminum sample.

Mixed rare earth oxides were added at the beginning of the experiments to simulate the influence of the europium oxide. They dissolved quickly and had no apparent influence on the nitric acid-aluminum reaction.

The reaction of aluminum with 1, 2, 4, and 6 M sodium hydroxide was investigated. The 6 M solution reacted most rapidly. In this case 20 minutes of very rapid reaction was followed by 33 minutes of slower reaction with a floating sample. The total time for complete dissolution

was 53 minutes. In the next fastest reaction, 4 M, 75 minutes was required for complete dissolution.

The acidification of the solution was carried out successfully by the rapid addition of concentrated nitric acid. The heat of the acid-base reaction caused the solution to boil and produced good mixing. When acid was added slowly a gummy precipitate of $\text{Al}(\text{OH})_3$ was first produced. This redissolved with some difficulty. Prolonged digestion might be required in the FPDL reaction tank where no agitation is available, if the procedure of slowly adding acid is followed.

The reagents consumed were as follows: 4.36 Kg of sodium hydroxide per Kg of aluminum and 17 liters of 15.5 M nitric acid per Kg of aluminum. The solution volume during dissolution was 18.2 liters per Kg of aluminum. The corresponding values per control plate are 28.8 Kg of sodium hydroxide and 112 liters of 15.5 M nitric acid. The total volume during dissolution is 120 liters.

Stainless steel test specimens were present during the dissolution tests. The weight losses were determined. The average loss was $1.5 \times 10^{-5} \text{ g/cm}^2$ during the acid dissolutions. The average weight loss during the caustic dissolution process was $1.9 \times 10^{-5} \text{ g/cm}^2$. All of this loss took place during the acidification portion of the process.

Each control plate will also contain 1726 g of tantalum. This metal does not dissolve in either of the processes investigated. Samples of tantalum sheet, $1 \times 1 \times 0.005 \text{ in.}$, were present during the dissolution tests. No attack of these samples was observed during the acid dissolution tests and only very slight attack was observed during the caustic dissolution tests. Further tests will be made of the dissolution of this material in caustic solution.

The gaseous products of both reactions could cause problems. The caustic reaction produces hydrogen gas which could cause an explosion hazard. The acid dissolution produces nitrogen oxides' NO and NO₂. A peak concentration of about 150 ppm could be reached in the Building 3039 off-gas stack during the initial fast part of the reaction. The nitrogen oxides could be removed by a scrubbing procedure or by a catalytic reactor similar to those in modern automobile exhaust systems.

The conditions for the reaction of the aluminum control plates with nitric acid were modified so that the reaction can be carried out in existing tankage. The reaction was to be carried out at an acid concentration and mercury catalyst concentration that would give a fast reaction leading to complete dissolution in about one hour. An acid concentration of 5.0 to 5.5 M and a mercury concentration of 0.4 g/liter gave these results. The temperature would be controlled by the boiling of the acid. The acid vapor would be condensed in a reflux condenser mounted on the tank and returned to the tank.

The reaction is initially rapid and exothermic. However, calculations indicated that the capacity of the reflux condenser would be exceeded several fold. Furthermore, the required volume of acid would occupy only a small fraction of the tank volume, and because the tank is heated only on the sides by a steam jacket, heating would be ineffective. Plans to heat the bottom surface with electrical heaters were abandoned because the high radiation background made work in this area infeasible. Both of these problems were solved by using a much larger volume of more dilute acid. A large fraction of the heat of reaction is used in heating the solution to boiling. When boiling does occur the reaction rate is well past the maximum and the reflux condenser is not overloaded.

First tests were carried out in beakers heated on a hot plate. Small additions of iron were made to the solutions to simulate the effect of an iron cable used to lower the control plates into the reaction tank. These tests established several facts:

1. The starting time and temperature of the reaction could be controlled by heating the aluminum and acid solution together and then adding the mercury catalyst. A rapid reaction started immediately at temperatures above 60°C . When the mercury was added before heating the starting temperature varied and was 20°C to 30°C higher.

2. The initial reaction was rapid but slowed greatly before the sample was completely consumed. The rate can be increased by adding more mercury. The addition of acid is less effective.

Because the heating of the solution by the reaction is a crucial part of the procedure, adequate tests could not be made in a beaker on a hot plate. The rate of heat loss per unit volume will be much lower for the large tank than for a 1 liter beaker. Consequently, tests were made in a Dewar flask. The heat loss per unit volume was estimated to be similar to that of the large tank if the water jacket was not used. The temperature of the reaction was measured by a thermometer passing through the stopper of the flask. Acid vapor passed through a tube into a graduated cylinder immersed in cold water. The condensate was collected and measured.

The results of one of these runs is given in Table 8.6.

Table 8.6. Reaction of Aluminum with Nitric Acid Under Nearly
Adiabatic Conditions^a

Time interval (minutes)	Temperature interval (°C)	Total rate of heat generation (Btu/hr)	Rate of heat absorption by condenser (Btu/hr)
0-3	64.2-66.2	1.9×10^5	Nil
3-5	66.2-69.2	3.0×10^5	Nil
5-10	69.3-78.5	3.8×10^5	Nil
10-20	78.5-92.2	2.8×10^5	Nil
20-22	92.2-95.2	3.1×10^5	Nil
22-24	95.2-98.2	3.1×10^5	Nil
24-30	98.2-102	2.3×10^5	8.8×10^4
30-33	102-102	1.8×10^5	1.7×10^5
33-35	102-102	1.2×10^5	1.0×10^5
35-40	102-102	1.2×10^5	1.1×10^5
40-45 ^b	102-102	1.0×10^5	8.0×10^4
45-50	102-101.4	6.6×10^4	5.1×10^4
50-55	101.4-101.9	8.6×10^4	5.1×10^4
55-60	101.9-102	2.9×10^4	3.3×10^4
60-75	102-101	1.7×10^4	1.5×10^4

^aOne hundred sixteen liters of 1.8 M nitric acid per Kg of aluminum. The solution also contained 0.88 g/liter of iron and 2.6 g/liter of mixed rare earth oxides. Time zero of the table is the time at which 0.36 g/liter mercury was added.

^b0.18 g/liter more mercury was added.

The rates of heat generation were calculated from the heat capacity of the solution and equipment and the temperature change and from the volume of condensate. No correction was made for the change in heat capacity of the solution as the temperature increased or for the change caused by the reaction of the aluminum and nitric acid. The heat of condensation was assumed to be 540 cal/g. Heat losses from the system were calculated using cooling rate data measured for the system after the completion of the reaction. The rates of heat production have been corrected to the conditions of the full scale reaction. It was assumed that the rate of heat production was proportional to the weight of aluminum present.

The maximum heat load of the reflux condenser is 1.7×10^5 Btu/hr. This is equal to the estimated capacity of the condenser. The conditions for the reaction are believed to be approximately correct for the full scale reaction. More exact information cannot be obtained by small scale laboratory experiment. A full scale test should be made if this reaction is used.

More work may be necessary concerning the off-gas. Other investigators report that H_2 , N_2 , N_2O , NO , and NO_2 are present in the off-gas. The situation is complicated further by the fact that NO_2 reacts with water to produce nitric acid and NO , and NO reacts with oxygen to produce NO_2 , but both reactions are comparatively slow. The maximum allowable NO_2 concentration in gas released to the atmosphere is 5 ppm. The NO will oxidize to NO_2 . The NO and NO_2 will be largely removed by oxidation and scrubbing. If no removal steps are taken and one mole of NO were produced per mole of aluminum, an average concentration of 21 ppm could be present in the off-gas leaving the stack for about 75 minutes once a week.

The average stoichiometry observed in the laboratory dissolutions was 3.81 moles of nitric acid per mole of aluminum. This is close to 3.75, which is the value for the reaction producing N_2O , but it is probably the average stoichiometry for several reactions that took place. The heat of reaction calculated from the data used in preparing Table 8.6 was 194 KCal/mole of aluminum. This falls between the values of 191.2 and 209.3 KCal/mole for the reactions producing N_2O and NO respectively.

The volume of solution used will be 768 liters. This will fit 43% of the volume of tank to be used for this process. This will be sufficient to allow effective heating using the steam jacket.

The acidity of the solution at the end of the reaction was 0.56 *M*, but in production the solution must be diluted to 6921 liters in order to achieve a sufficiently low level of radioactivity for handling in the waste disposal system. After dilution the acid concentration will be 0.06 *M*. This assumes a 3-1/2 year old control plate.

Several tests of the dissolution of tantalum metal in caustic solution were made. Sodium hydroxide concentrations of from 6 to 10 *M* were used at temperatures from 100 to 124°C. The weight losses were low; 0 to 1.9×10^{-4} g.cm⁻²hr⁻¹ were observed. The addition of sodium nitrite and sodium nitrate as oxidizing agents caused no increase. Previous tests in acid solutions had shown even less attack. No acceptable method for the dissolution of the tantalum present in the control plates has been found. This metal will accumulate in the bottom of the dissolution tank and can be flushed out with water.

The mercury catalyzed dissolution of control plates in nitric acid has been shown to be feasible and the approximate conditions for use of this

reaction in existing tankage have been established. More detailed work is not planned until it is decided that the process will be used.

Tests were made of the final steps of the process. The solution from a dissolution test was diluted and neutralized with sodium hydroxide, and the particle size distribution of the resulting precipitate was determined. The pH of the solution was measured after each of a series of dilutions, with results as given below.

Degree of dilution	pH
10 to 1	1.20
20 to 1	1.55
40 to 1	1.87
50 to 1	2.00

If this solution had been produced by the dissolution of a 3.5 year old plate, a 41-fold dilution would have been required to reduce the radionuclide content to 2 Ci/gal. The solution initially contained 7.9 g/liter of aluminum, 0.82 g/liter of iron, and 0.64 moles/liter of hydrogen ion.

The pH values are close to those predicted by simple dilution calculations. This indicates that the hydrolysis of the Al^{3+} and Fe^{3+} ions did not make a significant contribution to the acidity.

The addition of 37.5 milliliters of 1 M sodium hydroxide per liter of the diluted solution (50 to 1) caused precipitation and increased the pH to 9.9. The particle size distribution of this precipitate was determined using a Leeds and Northrup MICROTRAC. The results, given in Table 8.7 indicate that the precipitate could be pumped as a slurry.

Table 8.7. Particle Size Distribution of Precipitate

Particle size range (μm)	Percent of particles falling in range
16-22	4.2
11-16	12.5
7.8-11	22.2
5.5-7.8	24.5
3.9-5.5	18.9
2.8-3.9	11.7
<2.8	5.6

A study and estimate report for the facility for the HFIR control plate dissolution and other very high activity wastes was issued in August 1982 by the UCC-ND Engineering Division.

8.2.7.2 Electropolishing Decontamination Facility

An electropolisher and a vibratory finisher were installed at the Fission Product Development Laboratory (FPDL) to decontaminate metallic tools and small equipment for reuse at ORNL facilities or for salvage operations. The reduction in the volume of this type of material presently sent to the ORNL burial ground will aid in the conservation of the limited burial ground space available at ORNL.

The construction effort on the installation of the electropolisher was completed during the report period. ORNL personnel will complete the final tie-ins during the 1st quarter of FY 1983 and operational testing will be initiated in the 2nd quarter of FY 1983.

A vibratory finisher was purchased and installed during FY 1982 for decontamination work. Operational testing of this unit will commence in FY 1983.

Decontamination of contaminated equipment during FY 1982 was limited due to a lack of operating funds. Approximately 10,000 Kg of contaminated lead was cleaned to Health Physics tolerance for remelting, and several items of equipment were decontaminated for reuse in ORNL facilities.

8.2.7.3 Anaerobic Digestion of Cellulosic Materials and Animal Wastes

A preliminary process design and cost estimate has been made for a plant for volume reduction of low-level solid radioactive waste generated at ORNL. The process is based on extensions of existing anaerobic digestion technology and laboratory studies indicating the feasibility of this technology for digestion of the organic portion of low-level solid radioactive waste. The volume reduction factor for this process is expected to be approximately 20:1.

The waste to be treated consists of 10,500 ft³/year of uncompacted waste from ORNL and the Biology Division at Y-12. This waste is assumed to be segregated by the generators to exclude nondegradable materials. The waste is delivered to the plant site in dumpsters, where it is manually unloaded and fed to a wet shredder-pulper. Some sorting may be done at this point if necessary. Pulped waste is slurried in a mixing tank to the proper solids concentration for feeding to the digester (~1%). The 40,000 gallon digester is batch-fed daily with 11,600 liters water and 117 Kg waste. An equal amount of digest is removed to a surge tank. Hydraulic residence time in the digester is approximately two weeks, which is expected to produce 80% degradation of solids to soluble organics plus gas (CO₂ and CH₄). The gas will be vented (and flared if necessary) after passing through a demister. The digester is maintained at 35-40°C with

live steam and is mildly agitated. Caustic is added automatically to control the pH at 7, and small amounts of nutrients are added to maintain the microorganism population. Effluent from the digester is filtered to remove suspended solids and the supernatant is discharged to the ILW collection tanks and evaporator system (11,600 liters/day). The solids are dried and packaged for burial (23 Kg/day). The digester tank and associated equipment are housed in a prefabricated steel building. It is likely that the process can be operated more efficiently than the conservative basis used for study, which would allow some reductions in equipment size or increased capacity.

Capital cost and operating cost were estimated at the conceptual design level. The capital cost estimate includes 30% for engineering and an additional 40% of all itemized costs including engineering for contingencies. The summary of the capital cost estimate is:

<u>Item</u>	<u>\$</u>
Land improvements	35,000
Buildings	71,700
Special facilities	320,000
Outside facilities	42,600
Construction total	469,300
Engineering	140,700
Contingency	244,000
Total	854,000

The summary of the operating cost estimate is:

<u>Item</u>	<u>\$</u>	<u>\$/ft³</u>
Direct labor	56,000	5.33
Support labor and materials	11,000	1.10
Worked materials	6,600	0.63
Administrative and clerical	5,600	0.53
Duration materials	1,400	0.13
Total	80,600	7.72

These costs are comparable with costs for compaction, and are significantly less than the costs for incineration.

A small 70 liter digester has been ordered to continue process development work and to generate liquid and solid effluents for characterization and disposal studies in FY 1983.

8.2.8 Central Waste Disposal Facility Design and Development **(B. A. Hannaford, L. H. Stinton, and J. M. Chandler)**

This multiyear program is designed to provide a facility for disposal of low-level solid radioactive waste (LLSRW) for all three DOE Oak Ridge plants. The complex geology and hydrology of the DOE Oak Ridge reservation require that extensive site characterization work be carried out in order to qualify a site under DOE site selection criteria. A single large area on West Chestnut Ridge, near ORNL, was selected as the reference site. Preliminary field studies (geological, hydrological, and seismic refraction) were completed on six topographically attractive tracts. As a result of these studies, a decision was made to continue examining this site with more detailed field studies. Because the reference site could not yet be accepted with a high degree of confidence, however, a decision was made to select an additional, alternative site, on which to begin preliminary field studies.

For ORNL, the central waste facility will complement SWSA 7, the solid waste storage area being developed for low-volume, higher-activity solid waste.

8.2.8.1 Long-Range Planning for LLSRW Management Within the Oak Ridge Reservation

An analysis of the current status of Oak Ridge solid waste facilities was prepared for the UCC-ND Waste Disposal Steering Committee during FY 1982. The results of that study which pertain to low-level solid radioactive waste are summarized in Table 8.8. This showed that low-level radioactive waste represented the principal concern because only about 8 years capacity remained for this category of waste, and

**Table 8.8. Status of LLSRW Disposal Facilities
for DOE Oak Ridge Plants**

	ORNL	Y-12	ORGDP
Existing facility	SWSA 6	Bear Creek Valley	None ^a
Generation rate, ft ³ /yr	55,000	200,000	10,000
Regulatory status	Questionable ^b	Questionable ^{b,c}	N/A
Remaining capacity, yr	8-10 ^d	<10	N/A
Lead time to open new facility, yr	~8	~8	N/A

^aORGDP sends unclassified waste to Y-12 (α waste) and X-10 (β - γ waste) for disposal.

^bConcerned with proximity to water table.

^cQuestionable compliance with non-DOE regulations.

^dDevelopment of SWSA 7 to extend the capacity of SWSA 6 would add an additional 6-8 yr.

that about 8 years lead time is required in order to open a new facility. In addition, the existing ORNL and Y-12 burial grounds were judged to be in questionable compliance with regulations because of proximity of the waste to the water table.

For low-level radioactive waste, the study recommended the development of a central facility. The centralized concept was recommended on the basis that it minimizes the amount of land set aside for disposal, the site development costs (site characterization, pathways analysis, EIS, site monitoring), total capital costs, and operating costs — as compared with separate facilities for the three plants.

Another alternative which was discarded was commercial disposal. Commercial disposal costs were uncertain, but were likely to be in the range of \$20 to \$40 per cubic foot including costs of transportation, containers, and manifesting. A cost study by UCC-ND Engineering for disposal at a site on West Chestnut Ridge produced estimated costs ranging from about \$5 to \$10 per cubic foot, with no provisions for contingency, profit or closure allowances.

8.2.8.2 West Chestnut Ridge Site Characterization Studies

From a set of candidate sites which had been identified in a 1980 study, a single large area on West Chestnut Ridge was selected as the reference site for disposal of radioactive LLW. The favorable characteristics of the Knox residuum on Chestnut Ridge included its upland location, considerable depth to groundwater, and good soil thickness. Portions of the site which met the requirement for less than 20% slope were then identified for more detailed study (Fig. 8.21).

Fig. 8.21. Location and topography of the West Chestnut Ridge reference site for a central LLW disposal facility. Tracts within the dotted lines represent areas of Knox residuum having less than 20% slope; the largest tracts contain 90-100 acres. Directions to nearby DOE plants are indicated.

In terms of DOE siting criteria, the West Chestnut Ridge site characteristics which present the greatest uncertainty are (1) the potential geologic hazard constituted by the karst conditions and (2) the complexity of the site hydrogeologic system. The characteristics are interactive and both result from chemical weathering of the soluble dolomite bedrock.

8.2.8.2.1 Preliminary Investigations. Several investigations were carried out to characterize the site in a preliminary way. The investigations were centered on geologic techniques; supporting studies were done in hydrology and seismic refraction. These efforts provided the documentation on which to plan more detailed investigations which would evaluate the risk posed by the observed karst features.

The preliminary geologic investigations (8.8) on West Chestnut Ridge generally described the surface expression of karst features — roughly defined as potential sinkholes. These results were to provide part of the foundation for a later drilling program which would determine the approximate extent of subsurface karst features, such as solution cavities.

In order to assess the amount of overburden available for shallow-land burial, a seismic refraction survey (8.10) was carried out. Reported results showed that ample overburden exists on all six tracts. Seismograms for 16 locations showed that depth to bedrock varied from about 50 ft on lower slopes to over 120 ft near the higher ridges, with depths of about 75 ft or more generally available along the ridge crests.

Hydrogeological characterization of the West Chestnut Ridge site was carried out at two levels. Initial studies (8.8) was identified streams,

springs, and surface runoff features and noted the strong influence of the karst features on the hydrogeology. A subsequent hydrology study (8.12) employed temporary weirs at seven sites in order to measure surface water flow rates to be used as a basis for selection of permanent monitoring sites. Based upon preliminary results, a single location for a permanent flow monitor was recommended. Runoff from the eastern portion of the site was very much lower than from the western portion; a tentative recommendation was made that a second permanent monitor be installed in order to provide better water balance data for the eastern sub-catchments. Design of the first monitor was initiated; the design will employ a flume element in order to provide the desired rangability of approximately 1 lps to 4000 lps. Regular measurements of runoff were continued at the temporary stations.

8.2.8.2.2 Study Plan for Phase I Site Characterization

Investigation. A study plan (8.13) based on the preliminary site investigations was written to serve as the basis for more detailed studies of the site. These studies will be performed in a sequence with each phase based on the results of preceding work. The objectives are (1) to determine the three-dimensional extent of the soil zone above the water table and bedrock surfaces, (2) determine the existence of extensive bedrock solution cavities beneath the site, and (3) initiate the water balance study for the site.

8.2.8.2.3 Advisory Subcommittee Activities.

A Low-Level Waste Disposal Site Characterization Advisory Subcommittee was formed by T. H. Row, Chairman of the Waste Disposal Steering Committee (WDSC) for

UCC-ND. The Subcommittee, composed largely of geologists from ORGDP, ORNL and Y-12, was charged with advising WDSC on the use of the Knox formation on West Chestnut Ridge for LLW disposal. The Subcommittee monitored the field investigations, consulted with outside geologists and summarized their recommendations in an FY 1982 report (8.14).

8.2.8.3 Program Planning for the Central Facility

Planned activities for the new fiscal year generally reflected the results of site characterization reports (8.8 - 8.14) and called for carrying out a program of drilling, soil sampling, and hydrological studies on the more promising tracts of the West Chestnut Ridge site. Scoping pathways analysis would be performed and compared with the releases allowed under DOE 5820. An alternative site would be selected for preliminary investigation, as a hedge against the possible rejection of the West Chestnut Ridge site.

8.2.9 Metal Waste Management at ORO Sites (L. C. Williams)

Approximately 80,000 metric tons of uranium-contaminated scrap metal is currently stockpiled at USDOE uranium enrichment facilities in Ohio, Kentucky, and Tennessee. The program objective for the past year has been to make assessments of various disposal and reclamation options available for eliminating this scrap. The most promising process to date is size reduction followed by melting the metal, then casting the metal into block form for storage or use within the government. An additional benefit resulting from melting is that the bulk of the metal can be readily decontaminated during the smelting process. It is unlikely that commercial resale of the smelted metal will be permitted, but investigations have shown that an in-house need does exist for large quantities of metals in forms that can be easily produced. The current approach toward elimination of the contaminated metal scrapyards is by use of commercial operators for complete processing of the existing scrap metal at the various enrichment facility sites. Commercial operators will be contracted to do on-site segregation and size reduction of the metal. Smelting of the metal by commercial foundry firms may be completed either on- or off-site. The first phase of scrap processing will begin in early FY 1983 at the ORGDP, then expanded to the Paducah, Kentucky, and Portsmouth, Ohio, sites.

The major activities of the Metal Waste Management Program during FY 1982 were as follows:

8.2.9.1 Conceptual Design Requirements (CDR), Metal Smelting Facility

The CDR for the Metal Smelting Facility was completed by Lockwood-Greene Engineers, Inc. Construction, engineering, equipment, and installation costs were estimated at \$13M with an annual \$2.5M operating cost. The facility was designed for a through-put of 10,000 tons/y. Since the Metal Smelting Facility was not submitted as a FY 1984 line item, no follow-up action has been taken.

8.2.9.2 Evaluation of Disposal Options for the ORGDP Contaminated Scrap Metal

Fluor Engineers and Constructors, Inc., completed an evaluation of various disposal options for the ORGDP scrap. This study was completed in conjunction with the CDR for the Metal Smelting Facility. The disposal options investigated by Fluor included (1) burial of scrap "as is", (2) burial of the scrap with various modes of size reduction, (3) size reduction and shipment to a smelter facility at either Paducah, Kentucky, or Portsmouth, Ohio. Cost projections of these options are as follows:

<u>PROCESS</u>	<u>COST \$ PER/TON</u>
Burial "as is"	470
Burial - manual size reduction	600
Burial - semiautomated size reduction	575
Burial - fully automated size reduction	375
Fully automated size reduction, then shipment to Paducah or Portsmouth for final processing	350

The cost per ton shown includes capital equipment, operating, transportation, and burial ground costs.

8.2.9.3 Market Analysis for the Use of Smelted Steel

Foundry Design Company of Fort Lee, New Jersey, was contracted to perform a market analysis within the government to determine potential usage of steel produced from smelting of the contaminated scrap. This study revealed a potential need of 300,000 to 400,000 tons of steel for shielding purposes through 1992. The major user of the steel would be accelerator and fusion reactor facilities.

8.2.9.4 Commercial Interest for Processing Contaminated Scrap Metal

Foundry Design Company of Fort Lee, New Jersey, was contracted to investigate the commercial interest in processing the ORGDP scrap. The processing by a commercial firm would include on-site scrap sorting and size reduction, then smelting of the metal into block form that would be usable for shielding applications. Both on-site and off-site smelting of the scrap is currently being considered. Preliminary estimates indicate an average cost of \$55 per ton on-site sorting and size reduction. Smelting and casting blocks on-site is projected to run \$250 to \$300 per ton with off-site smelting estimated at \$450 to \$600 per ton. Contacts with the commercial firms will proceed during FY 1983 with a projected start of the sorting and size-reduction operations beginning during the third quarter of FY 1983.

8.2.10 Traffic Managers Support (D. S. Joy, L. B. Shappert, P. E. Johnson, C. S. Fore, and M. J. Fielden)

The major purpose of the Traffic Managers Support Program is to convert research and development transportation related computer programs into operational tools for the DOE traffic managers. The program was initiated in March 1982. The major phases of this project include Legislative Data Base modifications and application of the routing models in emergency response.

The scope of the Legislative Data Base has been expanded to include operational restrictions which apply to hazardous wastes in addition to radioactive materials. A new data base, Transportation of Hazardous Materials: Directory of Principal Contacts, was developed. By the end of FY 1982, this data base consisted of 568 entries from 32 different states designating those agencies (or personnel) that need to be contacted under particular conditions by a shipper prior to his movement of radioactive material. Also included are emergency response contacts throughout each state. Over 500 requests for information in the Directory of Principal contacts have been received and answered.

The HIGHWAY routing model data base, purchased initially in 1979, was expanded to include the location of principal DOE sites, nuclear reactor sites, and approximately 500 major airports. A separate airport data base was established containing runway information for the airports identified in the HIGHWAY data base which includes: runway length, runway width, runway composition, weight bearing capacity as a function of type of aircraft landing gear, runway lighting and instrument approach capabilities.

In order to respond to an incident involving radioactive materials, it may be necessary to transport trained personnel and/or specialized equipment to the site of the incident. An airport locator program was developed which, when queried, will list the airports closest to any site identified in the HIGHWAY routing model. Several constraints can be included in the search which would automatically exclude those airports not having the requested characteristics. The characteristics which can be specified include: minimum runway length, instrument landing capability, and weight of aircraft.

Draft reports of a users' manual for both the HIGHWAY routing model and the airport locator program were prepared and submitted for review.

8.3 PUBLICATIONS

J. D. Sease, E. M. King, J. H. Coobs, T. H. Row, "ORNL Radioactive Waste Operations," pp. 133-43 in Waste Management '82, Vol. II, Proceedings of the Symposium on Waste Management at Tucson, Arizona, March 8-11, 1982, ed. Roy G. Post, Arizona Board of Regents, 1982.

H. O. Weeren, J. H. Coobs, C. S. Haase, R. J. Sun, and T. Tamura. Disposal of Radioactive Wastes by Hydraulic Fracturing, ORNL/CF-81/245 (May 1982).

J. M. Chandler, R. P. Milford, B. D. Pietrzak, and S. P. du Mont A Comparison of Costs for Treatment and Storage or Disposal of Low-Level Solid Radioactive Wastes at ORNL, ORNL/TM-8092 (May 1982).

Evaluation Research Corp., Oak Ridge, TN, History of Disposal of Radioactive Wastes into the Ground at Oak Ridge National Laboratory, ORNL/CF-82/202 (June 1982).

B. M. Eisenhower, J. W. Oakes, J. H. Coobs, and D. W. Weeter, Current Waste Management Practices and Operations at Oak Ridge National Laboratory - 1982, ORNL-5917 (September 1982).

C. S. Haase, (in press) "Geological and petrological consideration relevant to the disposal of radioactive waste by hydraulic fracturing: An example of the U. S. Department of Energy's Oak Ridge National Laboratory," D. G. Brookins, ed. Proc. of the 6th International Symposium on the Scientific Basis for Nuclear Waste Management.

D. D. Huff, N. D. Farrow, and J. R. Jones, 1982. "Hydrologic Factors and ^{90}Sr Transport: A Case Study." Environ. Geol. Vol. 4(1):

D. D. Huff, 1982. "Hydrologic Factors and ^{90}Sr Transport of a Low-Level Waste Disposal Site." In R. G. Post (ed.) Proceedings of the Symposium on Waste Management at Tucson, Arizona. March 8-11, 1982. Vol. II, p. 349-356.

E. W. McDaniel, M. T. Morgan, J. G. Moore, H. E. Devaney, and L. R. Dole, Strontium Leachability of Hydrofracture Grouts for Sludge-Slurries, ORNL/TM-8198, Oak Ridge National Laboratory (March 1982.)

E. W. McDaniel, J. G. Moore, "Rheological Characterization of Cementitious Grouts Used to Dispose of Intermediate-Level Radioactive Waste by Hydrofracturing at Oak Ridge National Laboratory," Proceedings of the 4th Symposium on the Scientific Basis for Nuclear Waste Management, held at Boston, Massachusetts, November 16-19, 1981.

Lockwood-Greene Engineers, Final C. D. R. Metal Smelting Facility, Department of Energy, Oak Ridge, TN, Project Order 27, Task 01 (June 1982).

Flour Engineers and Constructors, Evaluation of Disposal Options for Contaminated Scrap Metal Located at the Oak Ridge Gaseous Diffusion Plant, ORNL/NFW-82/11 (June 1982).

Foundry Design Company, Market Analysis and Facility Modification, Metal Smelting Facility, Contract No. 62X-04174V (July 1982).

Foundry Design Company, Commercial Interest for Proceeding of Low-Contaminated Scrap Metal Located at the Oak Ridge, Tennessee, Uranium Enrichment Facility, Contract No. 12X-43121V (October 1982).

8.4 REFERENCES

- 8.1 A. M. Stueber, D. S. Huff, N. D. Farrow, J. R. Jones, and I. L. Munro, 1981. An Evaluation of Some ^{90}Sr Sources in the White Oak Creek Drainage Basin.
- 8.2 D. D. Huff, N. D. Farrow, and J. R. Jones, 1982. "Hydrologic Factors and ^{90}Sr Transport: A Case Study." Environ. Geol. Vol. 4(1).
- 8.3 A. M. Stueber, D. A. Webster, I. L. Munro, N. D. Farrow and T. G. Scott. 1981. An Investigation of Radionuclide Releases from Solid Waste Disposal Area 3, Oak Ridge National Laboratory. ORNL/TM-7323. Environmental Sciences Division Publication No. 1530. Oak Ridge National Laboratory, Oak Ridge, Tennessee. 53 pp. (Abstract): Geol. Soc. Am. Abstracts with Programs, Vol. 13, no. 1, p. 8.
- 8.4 J. A. Replogle, 1975. "Critical-Flow with Complex Cross Section." Proceedings of the ASCE Irrigation and Drainage Division Specialty Conference, August 13-15, 1975, Logan, Utah. p. 366-388.
- 8.5 K. O. Hasson, and C. S. Haase, 1982. "Stratigraphy of the Conasauga Group (Middle and Upper Cambrian) in the Valley and Ridge Province of East Tennessee" (Abstract): Geol. Soc. Am. Abstracts with Programs, Vol. 14, Nos. 1-2, p. 24.
- 8.6 J. Rodgers, 1953. Geologic map of East Tennessee with explanatory text: Tennessee Geol. Surv. Bull. V. 58, no. 2, 168p.
- 8.7 J. R. Markello, and J. F. Read, 1981. "Carbonate Ramp-to-Deeper Shale Shelf Transitions of an Upper Cambrian Intrashelf Basin, Nolichucky Formation, Southwest Virginia Appalachians: Sedimentology," Vol. 28, p. 57-597.5.

- 8.8 C. S. Haase, and N. D. Vaughan, 1981. "Stratigraphy and Lithology of the Conasauga Groups in the Vicinity of Oak Ridge, Tennessee," (Abstract) Geol. Soc. Am. Abstracts with Programs 13 (1):8.
- 8.9 W. D. Simpson, Principal Engineer. "Study and Estimate for Very High Activity Waste Handling Facility", UCC-ND Engineering Division, X-OE-191, July 30, 1982.
- 8.10 R. H. Kettelle, Report on Preliminary Site Characterization of the West Chestnut Ridge Site, ORNL/NFW-82/21 (November 1982).
- 8.11 W. P. Staub, Seismic Refraction Study of the West Chestnut Ridge Site, ORNL/NFW-82/26 (in preparation).
- 8.12 D. Huff, J. Elmore, and D. Farmer, Hydrologic Study and Evaluation of Ish Creek Watershed (West Chestnut Ridge Proposed Disposal Site). ORNL/NFW-82/27 (in preparation).
- 8.13 R. H. Ketelle, Study Plan for Phase I Site Characterization Investigation on the West Chestnut Ridge Site, ORNL/NFW-82/22 (in preparation).
- 8.14 W. E. Manrod III, Chairman, Low-Level Waste Disposal Site Characterization Advisory Subcommittee Report - Fiscal Year 1982, ORNL/NFW-82/28 (in preparation).

9. TRANSPORTATION PROGRAM

9.1 INTRODUCTION

The major objectives of the Defense Waste Transportation studies were addressed in a number of topics, including the legislative and regulatory information system, DOE support on the next international symposium on the topic of transportation, transportation criticality support activities, regionalized population assessments of highway routes, and TRUPACT testing support.

9.2 PROGRESS

Technical progress for each of the tasks of this program for this fiscal year just ended follows.

9.2.1 Defense Waste Transportation Studies (L. B. Shappert, D. S. Joy, G. E. Whitesides, C. S. Fore, and N. P. Knox)

The original Legislative and Regulatory information system was developed several years ago to maintain up-to-date information primarily on state and local legislative actions which could affect the movement of radioactive materials throughout the U.S. Recently the contents of the system were divided into a current and a historical data base, both of which were updated and maintained continuously throughout FY 1982.

Approximately 400 new bills were identified and status changes were made on hundreds more as the information was received. These two data bases were converted to a new data management system, INQUIRE, which will provide an improved means of data retrieval. Two summary papers were written and submitted to the Seventh International Symposium on Packaging and Transportation of Radioactive Materials (PATRAM '83). Work was initiated on compiling a draft users' manual for accessing the Legislative and Regulatory Information System. Over

200 requests for information from the system were received, processed, and answered.

The Seventh International Symposium on Packaging and Transportation of Radioactive Materials (PATRAM '83) has been organized under the sponsorship of TTC at Sandia National Laboratory and ORNL with the approval of DOE headquarters. All hotel arrangements have been completed and the meeting will be held in May 1983 in New Orleans, Louisiana. Authors' instructions for preparation of summaries and final papers have been developed by ORNL and approved by TTC.

In June 1979, under the aegis of the Nuclear Energy Agency (NEA) and its Committee on the Safety of Nuclear Installations (CSNI), a working group was formed at which member countries expressed interest in exchanging information and experience on various aspects of spent reactor fuel transportation, with the initial project to compare results of the application of criticality codes to a transportation problem.

To support this benchmarking of criticality codes on an international scale, a set of standard problems was developed which would permit the systematic evaluation of these codes which might be applied to shipments of spent light water reactor fuel in casks. An ORNL staff member chaired this effort to organize the problems and compare the results. Participating countries submitted results of the five main problems specified which were assembled and evaluated in a final report to members of the NEA.

The success of this exercise can be measured through the results obtained by applying various calculational methods to heavily shielded spent fuel casks containing a neutron poison and water coolant in the cask cavity. The studies concluded that the Monte Carlo and other transport methods can give satisfactory results for spent LWR fuel cask criticality calculations. The results also demonstrate that diffusion theory methods must be used with great care.

Review of the proposed revisions to the IAEA regulations for the Safe Transport of Radioactive Materials was also completed. The U.S. position concerning the adoption of the British "Q system" and the addition of a crush test were transmitted to the IAEA committee.

A computerized technique to evaluate the distribution of population densities along a transportation route were developed. Approximately 54 routes were generated to check the population density assumptions used in the RADTRAN program at Sandia National Laboratory. A special set of routes was also prepared showing the population density along both highway and rail routes which would likely be used in transporting waste to a repository site.

9.2.2 TRUPACT Transport System Tests (L. B. Shappert, W. D. Box, J. C. Rose, and J. O. Hylton)

The TRUPACT overpack is being designed by TTC at Sandia National Laboratory for the shipment of TRU wastes from storage locations to a repository. It is intended that the overpack withstand all tests required of type B packaging. In addition, it is desirable for waste packages carried inside the TRUPACT maintain their integrity under severe impact conditions. To develop information on the behavior of

these waste packages, the entire transport system (using a Supertiger overpack as a substitute for the TRUPACT overpack) was tested to accident conditions specified in the regulations. ORNL personnel helped load and seal the waste containers prior to transporting them to ORNL for testing. The transport system, weighing approximately 20,000 kg, was twice dropped from a height of 9 m at the TSF drop test facility. Accelerometer data recorded during the test were processed and plots of acceleration, velocity, and displacement versus time were produced at ORNL and transmitted to TTC.

9.3 PUBLICATIONS

C. S. Fore, Transportation of Radioactive Materials: The Legislative and Regulatory Information System, ORNL/TM-7439 (TTC-0280) (March 1982).

CSNI Working Group, Standard Problem Exercise on Criticality Codes for Spent Fuel Transport Containers (Final Report), CSNI Report No. 71, prepared for the Committee on the Safety of Nuclear Installations, Organization for Economic Cooperation and Development of the Nuclear Energy Agency, by Oak Ridge National Laboratory (G. E. Whitesides, ed.), Oak Ridge, TN (May 1982).

9.4 REFERENCES

None.

10. TRU WASTE MANAGEMENT PROGRAM

W. W. Pitt

10.1 INTRODUCTION

The Defense TRU Waste Management Program for FY 1982 consisted of four tasks. These tasks and their overall objectives were:

1. FUETAP Concrete Immobilization — determine the technological feasibility and applicability of autoclaved concretes for isolating TRU wastes.
2. Instrumentation Prototype — field test, calibrate, demonstrate, and evaluate a non-destructive assay system for TRU wastes.
3. TRU Waste Certification Program — develop plans and procedures for certifying ORNL TRU wastes.
4. Cementitious Waste Form Development for PREPP — provide assistance to EG&G Idaho in the development of a concrete formulation for stored TRU wastes.

10.2 PROGRESS

Technical progress for FY 1982 of the four tasks in this program follows.

10.2.1 FUETAP Concrete Immobilization* (L. R. Dole, G. C. Rogers, J. H. Kessler, T. M. Gilliam, and K. Duncan)

A large amount of alpha-contaminated waste material is generated in the preparation and fabrication of nuclear components. Much of this

*This project is jointly funded by Defense and Commercial Nuclear Waste Programs and is included in both sections of this annual report.

waste is combustible; the remainder may be metals, salts, sludges, and filters. Immobilizing these wastes in a FUETAP concrete offers advantages that include a waste form which is compatible with a wide spectrum of materials, has low leachability, and yields a dewatered final product with negligible radiolysis. Laboratory development and evaluation studies were conducted to obtain data for characterizing the technological feasibility and applicability of such concretes for isolating TRU wastes.

The activity in FY 1982 focused on: (1) PNL FUETAP samples, (2) an engineering scale FUETAP block, (3) Rocky Flats waste stream formulations for Building 374 and 741-2 sludges, and (4) leach testing of the 374 sludge.

10.2.1.1 PNL-FUETAP Samples

ORNL supplied PNL (W. Ross) with both cold and spiked FUETAP specimens, which are to be included in PNL's TRU-waste form evaluation program. Because of a time shortage, the samples were made with preliminary (but acceptable) formulas using simulated sludge provided by PNL. Time constraints did not allow for optimization of the formulas or for differences between simulated and real waste to be investigated.

10.2.1.2 Engineering Scale FUETAP Block

ORNL successfully cured and dewatered a 75-L FUETAP block, which had a porosity of only 14-volume percent, demonstrating this process on an engineering scale.

The 75-L FUETAP monolith (51 cm x 46 cm) containing 60 wt % SPI frit was dewatered September 29-30, 1981. This cylinder was dewatered in a

113.5-L (30 gal) drum with only one end exposed. The temperatures and pore pressures of the block were monitored at: (1) the center, (2) 1/2 R from the center and top, and (3) 1/2 R from the center and 2 R from the top.

The oven required 2 hours to reach 250°C and after 24 h the blocks temperatures at the measuring positions were: (1) 185°C, (2) 176°C, and (3) ~140-160°C. The latter was estimated because the thermocouple leads shorted out. No significant pore pressures were measured.

After 24 h in a 250°C oven, this 75-L monolith lost 5.3 wt % of water, as compared to 6.1 wt % loss from the same batch cast in a 4-L block. The initial water in the formula was 7.0 wt %, plus an additional 0.6 wt % from the frit. Therefore, 2.3 wt % water remained in the monolith. This is an acceptable water content and is comparable to the water content in the FUETAP concrete used in the α -radiolysis experiments. Therefore, no significant radiolysis is expected from FUETAPs with 2.3 wt % water.

10.2.1.3 Rocky Flats Waste Stream Formulations

A cement-based product (Table 10.1) has been developed which successfully incorporates 40 wt % Rocky Flats Building 374 waste with only a 20% volume increase. This formula's mix properties make it compatible with inexpensive, standard mixers. The curing and dewatering steps represent a fast, simple and low-temperature process. The compressive strength of this product is adequate and improves upon exposure to potential repository environments.

A simple, rapid, low-temperature solidification process for the Rocky Flats Building 374 waste was developed to achieve reduced processing costs and complexity. Previous experience at the ORNL/CCAC has

Table 10.1. Trial mix number 86

Component	wt %
Rocky Flats Building 374 Startup Sludge	40
Portland Type I Cement	22
Ottawa Sand	11.25
Reax-LP Water Reducer ^a	0.5
Fe ₂ O ₃	0.25
Water	24
Plastiment Water Reducer ^b	2

^aTrade name of Sika Chemical Corp.

^bTrade name of Westvaco, Inc.

shown that cementitious grouts can be rapidly cured and dewatered, using such simple, low-temperature processes. Based on this experience, a 24-hour cure at 90°C at 100% relative humidity followed immediately by a 24-hour dewatering at 250°C in dry air was chosen. For instance, the drums filled with the mixed grout could be put in a closed, 90°C water bath for 24 hours. After 24 hours, the water bath could be drained and the drums heated at 250°C for 24 hours. This would drive off most of the free water, which is not chemically bound in cementitious phases, reducing the drum weight and the consequences of α -radiolysis.

Compressive strength measurements were made using a Tinius-Jensen Universal Testing Machine, using a sample loading rate of 4.5 kg/s. A summary of the physical properties is found in Table 10.2.

An additional series of 5.1 cm cubes were used to determine the compressive strength changes after exposure to MCC-1 static leach

Table 10.2. Bulk Physical Properties of Trial Mix Number 86

Cured Density	: 1.75 g/cm ³
Dewatered Density	: 1.23 g/cm ³
Porosity	: 56%
Initial Water/Cement Ratio	: 1.1
Compressive Strength: After Curing	: 3.95 $\pm$ 0.16 MPa
Compressive Strength: After Dewatering	: 7.30 $\pm$ 0.24 MPa
Final Sample Volume vs. Initial Waste Volume:	~1.2/1

conditions. Four replicate specimens were leached in MCC-1 deionized water, silica water and brine for 28 days at 90°C. In this static leach test, the ratio of leachant volume to sample surface area was approximately 2.3/1.

Afterward leaching, these samples were dried at 250°C for 24 hours, cooled, and tested for compressive strength. As shown in Table 10.3, the

Table 10.3. Compressive Strength of Mix 86 Before and After Leaching in MCC-1 Leachants

Leaching treatment*	Compressive strength (MPa)	% Increase of compressive strength after leaching
No leaching	7.30 $\pm$ 0.24	-
28 days in 90°C deionized water	8.55 $\pm$ 1.88	17
28 days in 90°C silica water	8.24 $\pm$ 1.55	13
28 days in 90°C brine	12.32 $\pm$ 0.78	69

*Samples were cured at 90°C for 24 h and dewatered at 250°C for 24 h before leaching and dewatered at 250°C for 24 hr after leaching.

compressive strengths of these specimens increased after leaching. In the case of the MCC-1 brine, the compressive strength increased by 69%.

ORNL received and disassembled a drum of Rocky Flats 741-2 wastes. This heterogenous material contained the following visually identifiable phases and activities:

Drum fraction	Material	Activity $\mu\text{Ci/g}$
1/4	~Dry brown sugar + white crystals	100-200
1/2	~Moist brown sugar with 20 cm x 0.4 cm white blobs	100-200
1/4	~Peanut butter	50
Average		60

However, once this drum was broken down for glove box experiments funds were not available to do any formulation experiments.

10.2.1.4 Leach Studies of Cement-Based Waste Form with Rocky Flats Building 374 Sludge

Following the MCC-1 guidelines, 29 specimens of Rocky Flats Building 374 sludge in Mix 86 (see Table 10.1) were leached 3-, 7-, 14-, 28-, 56-, and 84-days in distilled water and MCC-1 standard brine at 90°C. The data were not reduced by the end of the fiscal year and no close-out funds were provided for FY 1983.

10.2.2 Instrumentation Prototype (F. N. Case, K. W. Haff, and F. J. Schultz)

10.2.2.1 TRU Waste Matrix and Isotope Effects Studies

Ten Neutron Interrogation Assay System (NIAS) radionuclide sources listed below were fabricated at ORNL and shipped to the Los Alamos National Laboratory.

<u>Radionuclide</u>	<u>Quantity</u>
Curium-244	1 m Ci
Americium-241	1 m Ci
Americium-243	1 m Ci
Plutonium-242	0.1 m Ci
Plutonium-240	2.5 mg
Plutonium-239	15 mg
Plutonium-238	10 m Ci
Neptunium-237	1.42 g
Uranium-233	20-100 mg
Uranium-235	20 mg

These sources along with sample drums of known matrices were used in the calibration and determination of sensitivities and detection limits for the neutron interrogation system. Moderator and absorber characteristics for the various matrices were also studied and correction factors obtained for fissile content assaying of the drums.

Neutron transmission measurements, which will be useful in elucidating waste matrix effects on both neutron absorption and moderation, were performed on 10 TRU waste drums. The transmission source used was a 0.14 μg ^{252}Cf sealed calibrated source.

Eight drums were removed from the TRU Waste Retrievable Storage Facility and shipped to LANL for use in NIAS system calibration and check out.

10.2.2.2 Differential Dieaway Neutron Interrogation Assay System

The LANL neutron interrogation assay system (NIAS) was received at ORNL on April 1, 1982 (see references 1 and 2 for details of the system).

The NIAS can perform the following operations:

1. total spontaneous fission neutron counting (passive interrogation).
2. neutron multiplicity counting for isotope identification (passive interrogation).
3. induced prompt-fission neutron counting for fissile mass assaying (active interrogation).

The present scheme for neutron assaying the TRU waste drums includes an active interrogation and a total coincidence neutron interrogation for all drums. If, after analysis of the data, any drum cannot be categorized properly a full neutron multiplicity measurement will be performed. Note that any drum exceeding 5×10^5 neutrons/second total neutron emission rate cannot undergo coincidence or multiplicity measurements in the present system.

An additional flux monitor (a low-pressure ^3He proportional counter) was positioned outside the neutron assay cavity opposite the neutron generator. This instrument well measures the neutron output of the generator to ensure that it is a constant value. The measured output varied less than 10%.

Throughput of drums in the active and total coincidence neutrons assay modes (not including neutron multiplicity measurements) was determined.

It was found that six to seven drums per hour could be assayed (40-50 drums per 8-hr shift) with sufficient personnel and equipment.

To date, 114 drums have been assayed by the NIAS. The distribution, as of October 1, 1982, of TRU waste drums by neutron emission rate is:

<u>Rate of Neutron Emission (n/s)</u>	<u>Drums</u>	<u>Total</u>
$<3 \times 10^3$	41	35.7
$3 \times 10^3 - 5 \times 10^4$	37	32.2
$5 \times 10^4 - 5 \times 10^5$	22	19.1
$>5 \times 10^5$	<u>14</u>	<u>13.0</u>
Total	114	100.0

The tentative distribution of TRU waste drums by activity level is:

<u>Activity Concentrations (nCi/g)</u>	<u>No. of Drums</u>
<10	9
$100 > \text{drums} > 10$	37
~ 100	13
>100	55

The drums listed as ~ 100 nCi/g may contain less than or greater than 100 nCi/g. Further isotopic investigation is required before a definitive answer can be given.

10.2.3 TRU Waste Certification Program (W. D. Box, J. D. Sease, E. M. King)

This was a one-year program and has been essentially completed. All pertinent procedures from the UCC Standard Practice Procedures Manual, the ORNL Health Physics Manual, the Radioactive Solids Waste Operation Manual, the Code of Federal Regulations, Quality Assurance Manual, and Divisional Building manuals have been assembled into an appendix to the Certification Program.

The first part of the ORNL Certification Program was informally submitted to WIPP in May and was reviewed with very few suggestions for change.

The portion remaining to be completed before final approval of the entire program is the addition of specific passages to the aforementioned procedure manual.

With the addition we hope to certify our first waste by the second quarter of FY 1983.

10.2.4. Cementitious Waste Form Development for PREPP (L. R. Dole and E. W. McDaniel)

During FY 1982 the following three subtasks were addressed: (1) waste form formulations studies, (2) INEL Processing Experimental Pilot Plant (PREPP) Proof-of-Principle Tests, and (3) review of the Ralph M. Parsons (RMP) Company conceptual design of PREPP selection of mixing equipment.

According to the PREPP flow sheet, the incinerator ashes will be separated into a fine and coarse fraction. The coarse material will be shredded metal and noncombustible debris, which will be dumped into 208-L (55 gal) drums. The fine ash fraction (<1.9 cm) will be mixed with cementitious materials, made into a fluid grout and poured over the drummed coarse debris. This grout is to flow into the interstices, encapsulating and fixing both the PREPP coarse and fine wastes into a durable monolith.

10.2.4.1 Formulation Studies

Workable (easily implaced and resists segregation) concrete formulations were prepared with both low noncombustible (high carbon content) and high noncombustible (low carbon content) simulated PREPP waste, as received from INEL. Four types of waste loadings were used: (1) High NonCombustible (HNC), screened (#20 NBS screen) to <850 micron particle size; (2) HNC random particle size; (3) Low NonCombustibles (LNC) screened to <850 micron particle size; and (4) Mixture of HNC and LNC screened to <850 micron particles. Data are summarized in Table 10.4. Using a ASTM I-II LA cement from the INEL area, workable concrete formulations were prepared with a lower (0.45) water/cement (w/c) ratio than

Table 10.4. Composition and properties of concrete that contain PREPP^a waste

Material	Amount (wt %)			
	1	2	3	4
Type I Portland Cement	22.55	18.70	16.74	21.56
ASTM Class F Fly Ash	19.61	16.26	14.56	18.74
D-65 Water Reducer ^b	0.98	0.81	1.46	0.94
H ₂ O	15.69	13.01	37.12	15.00
Waste: HNC (<850 micron)	41.18	34.15		39.36
HNC (large pieces)		17.07		
LNC (<850 micron)			30.57	4.40
Density (g/cm ³)	2.23	2.17	1.54	1.95
W/C (H ₂ O/cement)	0.70	0.70	2.22	0.696
7d Compressive Strength (MPa)	-	17.30	4.13	26.05
		(2508 psi)	(602 psi)	(3777 psi)

^aProcessing experimental pilot plant^bTrade name of Dowell Inc.

has been previously reported (0.7). The mix composition is listed in Table 10.5. This mix was thick enough to support three 2 cm pieces of high noncombustible waste, when they were placed on a 5.1 cm cube of fresh mix.

10.2.4.2 INEL PREPP Proof-of-Principle Tests

On May 20-21, 1982, a proof-of-principle test was conducted at INEL, EG&G, Idaho Falls, Idaho, to demonstrate the feasibility of encapsulating shredded metal with incinerator ash in a cement based grout.

Table 10.5. Composition and properties of concrete that contain PREPP^a waste

Material	Amount (wt %)
Type I-II LA Cement	23.67
ASTM Class F Fly Ash	21.62
Waste - High Noncombustable, <850 microns	42.20
D-65 ² H ₂ O Reducer ^b	0.87
Water	10.60
Density (g/cm ³)	2.23
W/C (H ₂ O/cement) ratio	.45
Slump	5-8 cm (2-3 in.)
14 d Compressive Strength	45.0 MPa (6521 psi)

^aProcessing experimental Pilot Plant.

^bTrade name of Dowell Inc.

The test was in conjunction with the PREPP to be built at INEL, Idaho Falls, Idaho.

Two basic grout formulas, one provided by RMP Company, design engineers for the PREPP facility, and an ORNL formula which contained fly ash and a water reducing agent, were considered for possible use. The RMP formula used no water reducing agent and used sand instead of fly ash. This formula is used in the construction industry and has a proven performance record when mixed with inert aggregate. The ORNL formula is similar, but tries to compensate for the uncertain effects of the incinerator ash. Each has a water/cement (w/c) ratio of about 0.70.

Different types of admixtures are added to concrete (or grout) to modify some of its physical and mechanical properties to make it more workable and/or more durable. A major function of the ORNL waste form development effort for PREPP was to select, based on past experience, admixtures that would be suitable for use in preparing grouts to fix PREPP waste and to verify performance by laboratory tests.

The first mixes were poured into 19-L (5 gal) cans containing a variety of >2 cm HNC particles. Grouts were vibrated around the waste using a construction "stinger" vibrator. In all cases, a slump of 13-15 cm (5-6 in.) was obtained, which is standard for pouring concrete. Five or six mixes were tested in this manner. Even though the grouts had an acceptable slump and no water collected on top, they did not have the right gel properties. Initial set had been achieved within 24 h. Final set was also probably achieved within 24 h (visually determined, not actually measured).

Grouts made with LNC ash tended to be water demanding and had a low slump and poor workability. These grouts required a w/c of approximately 2.0.

Samples made in 208-L (55 gal) drums were vibrated on a vibrating table. The drums were filled with scrap metal that had been heated to 77°C, in order to simulate incinerator conditions, and the grout containing the <2 cm incinerator ash was poured around the metal.

The RMP formula with approximately 35 wt % incinerator ash did not flow well around the metal and had false set. The grout was a dry crumbly material, which indicated a need for a wetting agent. ORNL recommended that D-65, a Dowell water reducer, be added. Past tests at

ORNL showed this admixture to be effective in separating the cement particles and suspending them uniformly in the mix water. This separation of cement particles resulted in a lubricating effect, as if there were more free water to decrease the grout viscosity. Unlike many admixtures, D-65 does not retard set time of the grout, and early set time is required for safest transfer in the PREPP process.

Twenty-five wt % D-65 was added to the mix, and the w/c was kept at approximately 0.70. The slump went to >30 cm (12 in.). The grout became extremely fluid and began to segregate. This mix was added to the first drum since two mixes were required to fill a 208-L (55 gal) drum.

Another mix was made to try to achieve a constant slump of 12.7 cm (5 in.). The dry solids were added to the mixer; D-65 was dissolved in minimum H₂O and added slowly as the solids were tumbled in the mixer. A w/c of 0.57 was required to achieve a constant slump of 12.7 cm (5 in.). This grout contained about 35 wt % incinerator ash. The grout was very plastic and workable. Visual tests showed enough air was entrained to provide good freeze-thaw property (This should be verified in the laboratory). A second mix using this formula was prepared to finish filling the second drum. This last grout possessed excellent rheological properties, and should pass the "proof-of-principle test."

10.2.4.3. Review of PREPP, the RMP Company Conceptual Design Selection of Mixing Equipment

Through an existing contract with EG&G, Idaho, ORNL was solicited to evaluate the RMP Company's Study Report for RWMC PREPP Cementing. After reviewing the RMP Company's report, ORNL chose to undertake an independent equipment review of relevant batch equipment. Basically, this

review defines and applies a well described evaluation procedure by establishing a systematic approach to the numerical scoring of selection criteria against specific performance requirements. It uses the RMP basis and scoring system with some modifications to normalize equipment comparisons and rationalize the numerical rating (1 to 10) of the performance criteria.

10.2.4.3.1. Evaluations. Of the nine performance categories listed by RMP, seven were rated by ORNL by assigning numerical weights (1 to 10) to each. The remaining two were evaluated and determined to have little impact on overall plant operating costs and therefore were not assigned a numerical weighting value. These categories along with their weighting factors are given in Table 10.6. The development status was given the highest weighting (8) and reflects the extent of operational experience with a particular model of mixer. Space requirements and controlled delivery to drums were also rated high because cell construction is the most important cost factor and delivery to the drums is one of the most critical and sensitive phases of the process. Tolerance to mix variation was given a weight of 5, because the fluctuations in the waste stream and their effects on mix rheology can be mitigated by formulation studies. The decontamination water recycle was given a rating of 6 because ideally the mixing operation should not result in a side stream that requires additional waste volume. The availability and ease of maintenance was given an 8 weighting factor (RMP rating = 6) because they are the keys to cost-effective operation throughout the plant life. The contamination confinement of the mixer was weighted four because other operations such as calcining and screening will determine the size

Table 10.6. Batch processing equipment rating sheet*

	Rotary non-tilt	Rotary tilt	Vertical side-out	Vertical bottom-out	Littleford horizontal
Development (8)	64	64	48	32	32
Mix variation tolerance (5)	50	50	20	20	50
Space required (8)	32	20	16*	16*	48
Contamination confinement (4)	24	8	40	40	32
Decontamination water recycle (6)	48	48	48	48	60
Availability/ maintenance (6)	48**	32**	48	48	48
Controlled delivery (8)	16	16	32	32	32
Total raw score (510 maximum)	282	238	252	236	302

*ORNL's analysis dropped RMP's equipment cost (3) and power consumption categories and raised the weighted availability and maintenance from (6) to 8.

**Normalized to 10 drums/batch or 20 drums/day - continuous.

***High maintenance doses from contamination.

of the off-gas cleanup system. ORNL dropped the power consumption category because there is little practical difference between mixers, and this will be an insignificant fraction of the plant load. The equipment costs were also deleted because they were not considered an important impact on the overall plant costs.

The equipment listed on the RMP report and a Littleford mixer Model KFM 3000D were evaluated in the ORNL study. The results are listed in Table 10.6.

10.2.4.3.2. Conclusions. Table 10.6 shows ORNL's analysis without RMP's categories of equipment cost and power consumption and with raising the weight of the availability and ease of maintenance from six to eight. This analysis shows about a 10% difference between the Vertical Sideout (252), the Rotary Non-Tilt (282) and the Littleford horizontal (302) mixers. The average score of these three is 280 with a spread of about +28 points. These ratings are close enough to be inconclusive and the relative total scores could be reordered with the development of more accurate evaluations of maintenance requirements, space requirements, and controlled delivery systems. Resources did not allow ORNL to pursue these critical items in sufficient detail.

10.3 PUBLICATIONS

None.

10.4 REFERENCES

None.

COMMERCIAL NUCLEAR WASTE MANAGEMENT PROGRAMS

6. INTEGRATED DATA BASE PROGRAM

K. J. Notz

6.1 INTRODUCTION

This program was initiated in FY 1980 for the purpose of establishing and maintaining a computerized, integrated data base of current and projected nuclear wastes in the United States. A consistent set of evaluated information on waste inventories, characteristics, sources, package descriptions, and locations of interim storage and disposal sites is needed by the DOE and other agencies and contractors for use in waste program planning and systems analysis. The data base will also include costs, and will be capable of taking into account the effects of different methods of chemical treatment and packaging on the characteristics of final waste forms.

6.2 PROGRESS

Technical progress for each of the tasks of this program for this fiscal year just ended follows.

6.2.1 Integrated Data Base Program (C. W. Alexander, W. L. Carter, C. W. Forsberg, H. W. Godbee, B. M. Horwedel, A. H. Kibbey, and G. W. Morrison)

This program is jointly funded by Commercial Waste Management and Defense Waste Management, and is carried out in a fully integrated manner. See Section 12 for the Progress, Publications, and References dealing with this program.

6.3 PUBLICATIONS

See Section 12.3.

6.4 REFERENCES

See Section 12.4.

11. HIGH-LEVEL WASTE MANAGEMENT PROGRAM

W. W. Pitt

11.1 INTRODUCTION

The Commercial High-Level Waste Management Program for FY 1982 consisted of three tasks. These tasks and these objectives were as follows:

1. ORNL Fixation of Wastes in Concrete — prepare an evaluation report on FUETAP concrete as an alternative waste form for high-level wastes.
2. High-Level Waste Container Development — conduct final examination of HLW canister material and the document study.
3. ORNL Sol-Gel Coated Particle Waste Form Development — document results of work done on sol-gel coated particle waste form for high-level wastes.

11.2 PROGRESS

Technical progress for each of the tasks of this program for FY 1982 follows:

11.2.1 ORNL Fixation of Waste in Concrete (L. R. Dole, J. H. Kessler, M. T. Morgan, and G. C. Rogers)

This task was performed as part of the DOE program to evaluate alternative processes to solidify high-level radioactive wastes and to evaluate the quality of the solid waste product. Over the past year, studies have been directed toward characterizing the technological feasibility and applicability of concrete. In particular, FUETAP concretes (Figure 1) have been developed as alternative waste forms for both defense and commercial radioactive wastes. This program was terminated at the end of

FUETAP FLOW SHEET

Fig. 11.1. Flow sheet for FUETAP concrete developed for the Savannah River Plant high-level defense wastes.

FY 1982 with the preparation of a draft evaluation report, which will be published as ORNL/TM-8579.

The following sections summarize selected topics from this evaluation report.

- (1) Processability
- (2) Radiolytic Stability
- (3) Leachability
- (4) Thermal Stability
- (5) Summary and Conclusions

11.2.1.1 Processability of Concrete Waste Hosts

The processability of FUETAP concretes is a major advantage of this radioactive waste disposal alternative. For example, one of the alternative flow sheets for the SRP defense high-level waste requires that two tons of dry solids be added to a full charge of water and set retarder in a ribbon mixer over two h. To assess the consequences of a broken blender-shaft or a frozen transfer boom, the threshold solids for set, the phase separation and the set-time were investigated.

Shown below are the threshold solids addition for set, the phase separation and the set-time for the MCC-1 FUETAP formulation with simulated SRP-HLW defense waste.

These results show that for this process step, no auxiliary agitation, backup transfer system, or extraordinary operational procedures are required to deal with abnormal events.

11.2.1.2 Radiolytic Stability

The long-term α -radiolysis experiments at 25-30°C used 10 ml carbon steel capsules, containing FUETAP concrete specimens (1.5 ϕ cm x 3.5 cm;

<u>Percent of total solids added</u>	<u>Set-time</u>	<u>Percent phase separation</u>	<u>Volume of liquid</u>	<u>Volume of solids</u>
25	No set	43	117.0 cm ³	155.0 cm ³
37.5	"	31	91.0	200.0
50	"	14	56.0	340.0
62.5	"	<1	3.0	365.0
75	<10 h	0	-	399.0
87.5	4-10 h	0	-	412.0

6.2 cc; 8.2 g solids) which were spiked with 4.5 mg (tests 1, 2, 3), 0.5 mg (tests 4, 5) and 1.5 mg (test 6) of curium -244 (12.3 mg per curie). The dead volume (~5.8 cc) of each test cell was measured by expanding a known volume and pressure of argon into the test capsules. The test's experimental conditions and are summarized below.

<u>Dewatered 250°C, 24 h</u>	<u>Dose rate M rad/hr</u>	<u>G(total gas) molec./100 eV</u>	<u>Dose range for average rad</u>
No	.56	0.095	.1-7x10 ⁸
Yes	.56	0.006	1-2x10 ⁹
Yes	.56	0.005	.01-2x10 ⁹
Yes	.063	0.001	3.7x10 ⁸
Yes	.19	0.002	.2-1x10 ⁹
Yes	.19	0.008	1-5x10 ⁸

The conservative total gas G (Molecule/100 ev) values were taken from the averages of the steeper initial linear pressure rises. These G (total) values are a fifth to a third of Bibler's values.¹

Use of these G values is over conservative, since they were taken from the initial steep linear portion of the curves. If they had been taken from the flatter region of the Pressure vs. Time curves (Figure 2), the calculated integrated off-gasing would be lower by 3 to 4 fold. Furthermore, actual waste particles would be of sufficient size to reduce the dose to the concrete matrix two orders of magnitude compared to these specimens which were spiked using a nitric acid solution, giving a homogeneous distribution of Cm-244.

Independent studies have shown that FUETAP concretes can be very effective recombination catalysts. But, these small (~5cc) samples may have had insufficient internal diffusion paths for effective recombination; and therefore, there may be a significant sample geometry effect. It is possible that radiolysis measurements on very small specimens could be conservatively high by factors of 10-30X, due to the high escape surface-to-volume ratio.

The combination of these factors could result in an overestimate of radiolysis rates of 3-12,000X, if the total gas G-values are used in a simple linear model.

Figure 11.1 shows estimates of gas pressurization for non-dewatered and dewatered FUETAP concretes in a perfectly sealed 200 L drum filled to 90% capacity. These estimates are made from actual data points and did not use a linear extrapolation of the initial G-values.

11.2.1.3 Leachability

Using the MCC-1 static leach tests, two FUETAP formulation series, MIX:MCC-1 and MIX:MCC-6, were evaluated for the following effects:

- a. The effect of substituting slag and fondu cements for Portland Type I Cement (Table 11.1).

Fig. 11.2. Gas generation in FUE-TAP concrete.

- b. The effect of adding sodium bentonite and Indian Red Pottery clay (Table 11.2 and Figure 11.3).
- c. The effects of MCC standard leachants and temperature on Cs, Sr, U, Ce, Nd from a FUETAP MIX: MCC-6 (Table 11.3).

These data (Table 11.1) show that the FUETAP process results in durable product with three readily available commercial cements.

Table 11.2 and Figure 11.3 demonstrate that clay additives reduce the initial Cs washout and leach rates by about a factor of two.

This effect of initial washout on the reported MCC-1 leach rates is illustrated in Figure 11.4 using MCC-1 90°C strontium leach data. In these cases, the short-term leach rate data reported, as specified in the MCC-1 testing procedure, are dominated by washout and are therefore biased. Because the MCC-1 procedure results in testing specimens with a sample surface-to-volume (s/v) ratio of ~7 and the actual waste forms have a s/v < 0.1, the MCC-1 leach data implies nearly a 100x larger fractional release rate than actually expected.

Using a revised simulated SRP reference waste and the same FUETAP formula, the FUETAP mix: MCC-6 series was leached to determine the effects of the three standard MCC-1 leachants and two temperatures, 40°C and 90°C (Table 11.3). Comparing the data in Table 11.3, only the leaching of Sr and U in silica water shows a statistically significant difference between the MCC-1 leach rates (solubilities) at 40 and 90°C. Since the revised SRP simulated waste used in this FUETAP mix: MCC-6 was more water demanding than the previous reference waste (Table 11.1 and 11.2), this formulation resulted in a specimen with 50% greater porosity (~40%). As a result, the Cs washouts were higher (4-5x) and the 28-day

Table 11.1. Effect of cement type on the leach rates of various elements from FUETAP (MCC-1) concrete in deionized water

Cement type	28-Day MCC-1 leach rates at 90°C			$\frac{g_1}{M^2 - \text{day}}$
	Al	Fe	Mn	Sr
Type I Portland	3.31	0.0041	0.00096	0.43
Type I-S Slag	1.57	0.0020	0.0234	0.39
Fondu (high alumina)	5.67	0.0091	0.0152	0.83
50% Type I + 50% Fondu	4.11	0.0159	0.0268	0.11

g_1 = reduced grams = g-element/mass fraction in the waste.

Table 11.2. Effect of Indian Red Pottery plus bentonite clay additives on the leach rates of cesium from FUETAP (MCC-1) concrete

Leach time (days)	MCC-1 leach rates at 90°C				$\frac{g_1}{M^2 - \text{day}}$
	DIW leachant		SIW leachant		
	*Clays added	None added	*Clays added	None added	
3	97.0	150.0	63.0	99.0	
7	53.0	81.0	30.0	52.0	
14	35.0	44.0	19.0	35.0	
28	-	29.0	-	16.0	

DIW - Quartz distilled water

SIW - MCC silica water

*3.5% Indian Red plus 3.5% Bentonite added

ORNL-DWG 81-15760

Fig. 11.3. Effects of clay additives on cesium leach rates from FUE-TAP (MCC-1) in silica water at 90°C.

ORNL DWG 81-598

Fig. 11.4. FUETAP with Type 1 Portland Cement and SRP-4 — simulated waste MCC-1 strontium leach results at 90°C in deionized water.

Table 11.3. Results of 40 and 90°C MCC-1 Cs, Sr, U, Ce and Nd 28-day
leach tests of FUETAP concrete mix: MCC-6 with Savannah River
Plant (SRP) simulated waste formula: SRP6A with Indian Red
Pottery clay (reduced-grams/M²·d)

	Temperature (°C)	Cs	Sr	U	Ce	Nd
Distilled water (DIW)	90	48 ± 7	.27 ± .02	.035 ± .008	.11 ± .02	<.1
Silica water (SIW)	40	46 ± 3	1.3 ± .1	.007 ± .003	.02 ± .02	<.1
	90	37 ± 3	.30 ± .01	.02 ± .01	.02 ± .01	<.1
Salt brine (BRI)	40	59 ± 12	17 ± 6	.06 ± .01	<.1	1.0 ± .5
	90	53 ± 5	23 ± 5	.06 ± .01	<.1	.6 ± .2

leach rates are 0.8x and 3x higher than the DIW and SIW 28-day leach rates in Table 11.2, because of the washout bias in these data.

A statistical analysis of 70 MCC-1 leach results for uranium (Tables 11.4-11.7) shows that there was significant difference in uranium concentrations (ppm) between simulated waste formulas SRP-6A and SRP-6B, which had 3.5% Indian Red Pottery clay and sodium bentonite, respectively (Table 11.4, while Table 11.5 shows that there was no significant time dependence of the uranium concentrations (ppm). This indicates that the MCC-1 test does not measure U-leaching, but rather U-solubility. Table 11.6 shows that it is the same for 40° and 90°C in WIPP Brine B (BRI). On the other hand, there was an increase in U-solubility with temperature in the case of MCC-1 silica water (SIW).

No statistically significant difference in uranium solubility between dionized water (DIW) and the silica-water (SIW) leachants was observed (Table 11.7). However, the WIPP-brine B (BRI) could hold more uranium in solution, probably due to the lower final pH of these brine leachates.

11.2.1.4 Thermal Stability

Thermal expansion measurements were made using samples whose initial size was approximately 15 mm diameter by 31 mm long. These samples were made using the MCC-1 formulation with SRP-4 waste solids cured and dewatered by the standard FUETAP method (100°C - cure, 250°C - dewater, 24 hours).

Samples were heated at a rate of 4°C per minute to 800°C, held at 800°C for approximately five minutes, then returned to room temperature at a rate of 4°C per minute.

Table 11.4. Duncan's multiple range test* for variable:
uranium concentration vs. waste formula

(Means with the same letter are not significantly different.)

Alpha = 0.05 DF = 66 MSE = .0244771
Harmonic mean of cell sizes = 34.5429

Duncan grouping	Mean	N	Waste formula
A	0.16027	39	SRP6B
B	0.03877	31	SRP6A

Table 11.5. Duncan's multiple range test* for variable:
uranium concentration vs. days leached

(Means with the same letter are not significantly different.)

Alpha = 0.05 DF = 66 MSE = .0244771
Harmonic mean of cell sizes = 5.69637

Duncan grouping	Mean	N	Day
A	0.15214	10	3
A	0.10635	53	28
A	0.04583	3	7
A	0.03932	4	14

*Note: This test controls error rates at different levels depending on the number of means between each pair being compared. Its operating characteristics somewhat resemble Fisher's unprotected LSD test.

Table 11.6. Duncan's multiple range test* for variable:
uranium concentration (ppm) vs. temperature

(Means with the same letter are not significantly different.)

On waste formula SRP6A leached in MCC-1 brine

Alpha = 0.05 DF = 5 MSE = 2.9E-04

Harmonic mean of cell sizes = 3.42857

Duncan grouping	Mean	N	Temperature °C
A	0.076650	4	90
A	0.171767	3	40
<p>On waste formula SRP6A leached in MCC-1 silica water</p> <p>Alpha = 0.05 DF = 11 MSE = 2.1E-04</p> <p>Harmonic mean of cell sizes = 6.46154</p>			
A	0.032529	7	90
B	0.011850	6	40

Table 11.7. Duncan's multiple range test* for variable:
uranium concentration vs. leachant type

(Means with the same letter are not significantly different.)

Alpha = 0.05 DF = 66 MSE = .0244771

Harmonic mean of cell sizes = 23.1555

Duncan grouping	Mean	N	MCC-1 leachant
A	0.24468	21	Brine
B	0.060207	23	D-Water
B	0.03576	26	Silica water

*Note: This test controls error rates at different levels depending on the number of means between each pair being compared. Its operating characteristics somewhat resemble Fisher's unprotected LSD test.

A typical expansion curve for the MCC-1 FUETAP formula is shown in Figure 11.5. During the heating cycle the sample length increased almost linearly to approximately 0.2% at 400°C. The sample length continued to increase in length up to approximately 580°C. This was caused by the transition from low to high quartz. The sample length remained fairly constant between 600°C and 700°C. At approximately 700°C the sample began shrinking rapidly. At 800°C the sample expansion was approximately -0.7%.

Upon cooling, the sample shrank linearly until a sharp drop occurred at approximately 580°C, caused by the reversible transition from high to low quartz. At room temperature again, the final thermal expansion was approximately -1.7% of the original length.

Using the expansion curve in Figure 11.5, an average thermal expansion coefficient of $4.7 \times 10^{-6}/^{\circ}\text{C}$ can be derived between 20°C and 400°C. This coefficient is about half that of most HLW glasses and hard rolled steel.

11.2.1.5 Summary and Conclusions

Where do FUETAP concretes stand with regard to the major questions concerning the performance of radioactive waste forms?

With regard to leachability, the dynamic leach rates of transuranics are below our detectable limits ($<0.01 \text{ g}_1/\text{M}^2\text{d}$). Cs and Sr leachabilities compare with glass, but the initial washout is higher. These leach rates are sufficiently low to meet the intent of the proposed NRC 10 CFR 60, without credit for the package.

Radiolytic decomposition of these low-water ($<2\%$) FUETAP concretes is negligible over 7×10^5 years. They are thermally stable to 900°C and have thermal expansions less than steel.

ORNL DWG 81-547R

Fig. 11.5. Typical expansion curve of an MCC-1 type concrete.

FUETAP concretes are high strength concretes (40–100 MPa, compressive strength) and produce less than 0.4 wt % respirable fines upon impact of $\sim 10 \text{ J/M}^3$. Their porosities, permeabilities ($\sim 10^{-5}$ darcy), thermal conductivities, and densities are inversely proportional to their water:cement (w/c) ratios. In turn, the w/c depends on the process rheological limits and the water demand of the specific waste stream. The formula must be tailored to obtain compatibility with specific waste stream, processing/implacement technology, and disposal environment.

Since concrete is already the principle radwaste host for most low- and intermediate-level waste, and accelerated curing is a common industry practice, no inventions are required to apply the FUETAP process. It uses inexpensive and readily available materials, which process at low temperatures. This process works with a broad range of waste streams and cementitious materials.

The cementitious and radioactive waste phases are complex mixtures, and the identification and location of their specific nuclides has just begun. However, FUETAP concretes have already been shown to work as durable, tenacious hosts for most of the major defense and reprocessing radioactive waste streams.

11.2.2 High-Level Waste Container Development

The work accomplished under this program in FY 1982 included the completion of the post exposure examination of test specimens, and preparation of a final report. The abstract of the final report is given below:

H. E. McCoy, Studies of Waste-Canister Compatibility, ORNL/TM-8491 (in publication).

Compatibility studies were conducted between 7 waste forms and 15 potential canister structural materials. The waste forms were Al-Si and with carbon or carbon plus silicon carbide. The canister materials included carbon steel (bare and with chromium or nickel coatings), copper, Inconel, Cu-35% Ni, titanium (grades 2 and 12), several Inconels, aluminum alloy 5052, and two stainless steels. Tests of either 6888 or 8821 h were conducted at 100 and 300°C, which bracket the low and high limits expected during storage. Glass and FUETAP evolved sulfur, which reacted preferentially with copper, nickel, and alloys of these metals. The Pb-Sn matrix alloy stuck to all samples at 300°C, but the extent of chemical reaction was not determined. The carbon-coated particles also stuck to most samples at 300°C, but the extent of chemical reaction was not determined.

Testing for 0.5 h at 800°C was included because it is representative of a transportation accident and is required of casks containing nuclear materials. During these tests (1) glass and FUETAP evolved sulfur, (2) FUETAP evolved large amounts of gas, (3) Synroc stuck to titanium alloys (4) glass was molten, and (5) both matrix alloys were molten with considerable chemical interactions with many of the canister samples. If this test condition were imposed on waste canisters, it would be design limiting in many waste storage concepts.

11.2.3 ORNL Sol-Gel Coated Particle Waste Form Development

Progress during FY 1982 included completion of several development tasks initiated during FY 1981, and final documentation of the program. The documentation was completed in four reports. Abstracts of the reports are included below:

W. D. Arnold et al, "Preparation of Gel Spheres Containing Simulated High-Level Radioactive Waste," Radioactive Waste Management and the Nuclear Fuel Cycle, Volume 3 (1), September 1982 (pp. 57074).

Gel-derived microspheres containing simulated high-level radioactive waste sludge and ceramic additive materials were prepared by internal gelation at waste loadings as high as 90%. The gel spheres were amenable to subsequent drying, sintering, and coating procedures producing highly leach-resistant crystalline waste forms suitable for permanent disposal. Potential application of this technique to commercial power reactor waste processing was demonstrated by incorporating simulated Purex solvent extraction waste in gel spheres at 20% waste loading.

W. D. Arnold et al, Incorporation of Simulated High-Level Nuclear Waste in Gel Sphere, ORNL-TM/8124, December 1982.

Gel sphere technology developed for reactor fuel fabrication was applied to the fixation of simulated high-level radioactive waste in crystalline ceramic form for permanent disposal. Gel spheres containing simulated alkaline defense waste sludges and ceramic matrix materials were prepared by internal gelation at waste loadings as high as 90%. The gel spheres were amenable to subsequent drying, sintering, and coating procedures to produce crystalline waste forms with extremely high leach resistances. Potential application of this technique to the processing of commercial power reactor waste was demonstrated by incorporating simulated Purex solvent extraction waste in gel spheres with up to 20% waste loading. Cesium present in the simulated waste was adsorbed on zeolite and immobilized by coating with carbon.

D. P. Stinton et al, "Coating of Crystalline Nuclear Waste Forms to Improve Inertness," Journal of the American Ceramic Society, Vol. 65, No. 8.

Microspheres of a crystalline waste form prepared by sol-gel processing were successfully coated with layers of pyrolytic carbon and silicon carbide to isolate the radioactive wastes from the biosphere. A separate process for cesium immobilization was developed, which loads 5 wt % Cs onto zeolite particles for subsequent coating. Pyrolytic carbon-coated particles showed leach rates ~2 to 4 orders of magnitude less than the candidate reference borosilicate glass waste form. Aqueous leach-test results of coated waste forms were below detection limits of such sensitive analytical techniques as atomic absorption and inductively coupled plasma atomic emission.

D. P. Stinton et al, "Immobilization of Radioactive Cesium in Pyrolytic Carbon-Coated Zeolite," submitted to the Journal of the American Ceramic Society in September 1982.

All waste forms currently being developed for radwaste disposal have difficulty immobilizing cesium. A new waste form was developed to dispose of cesium by loading up to 5 wt % Cs onto zeolite granules. These zeolite particles are then heat-treated and coated with pyrolytic carbon and/or SiC to completely immobilize cesium. The leach rate of cesium from pyrocarbon-coated zeolite is 3 to 4 orders of magnitude less than for other waste forms. We therefore propose that cesium be removed from the primary waste stream and loaded onto zeolite granules for subsequent coating. The total waste form (primary waste form plus coated zeolite) would have acceptable leach resistance for all radwaste components and extremely leach resistant for cesium.

11.4 REFERENCES

1. N. E. Bibler in: Proceedings of the Conference on the Scientific Basis for Nuclear Waste Management, Vol. 2, C. J. M. Northrup ed. (Plenum Press, New York 1980), pp. 525-582.
2. S. Katz in: Proceedings of the Conference on the Scientific Basis for Nuclear Waste Management, Vol. 2, J. G. Moore ed. (Plenum Press, New York 1980), pp. 577-584.

12. INTEGRATED DATA BASE PROGRAM

K. J. Notz

12.1 INTRODUCTION

This program was initiated in FY 1980 for the purpose of establishing and maintaining a computerized, integrated data base of current and projected nuclear wastes in the United States. A consistent set of evaluated information on waste inventories, characteristics, sources, package descriptions, and locations of interim storage and disposal sites is needed by the DOE and other agencies and contractors for use in waste program planning and systems analysis. The data base will also include costs, and will be capable of taking into account the effects of different methods of chemical treatment and packaging on the characteristics of final waste forms.

This program is jointly funded by Commercial Waste Management and Defense Waste Management, and is carried out in a fully integrated manner. Thus, this work is shown under both funding sources (see also Section 6) but is reported in detail only once, in this section.

12.2 PROGRESS

Technical progress for the task of this program for the fiscal year just ended follows.

12.2.1 Integrated Data Base Program (C. W. Alexander, W. L. Carter, C. W. Forsberg, H. W. Godbee, B. M. Horwedel, A. H. Kibbey, and G. W. Morrison)

The major thrust of the program is to develop and build a reliable, easily accessible, and versatile data base that covers all domestic spent fuel and radwaste and that uses computerized data handling capabilities effectively.

Essential components of the data handling capability are (1) computerized data input, (2) electronic transfer of data, and (3) a data base management system that provides direct, easy, and fast access to data in the system. This access must also allow tailored data output to serve specific needs in terms of content, format, and presentation, both tabular and graphical. In support of these objectives, the IDB Program has major tasks in four areas:

- To develop the data handling capability, using state-of-the-art hardware and software, and focusing on the urgent need to facilitate fast, accurate, and easy access by users;
- To develop a data base of inventories, characteristics, source terms, packaging, and costs covering all domestic spent fuel and radwastes (including HLW, TRU waste, LLW, airborne waste, mill tailings, remedial action wastes, and D&D wastes) from the commercial fuel cycle, from institutional and industrial (I/I) activities, and from DOE/defense functions;
- To develop the necessary models to allow projections of future quantities and characteristics of spent fuel and radwaste, including the capability to deal easily and quickly with alternative projection scenarios and treatment/handling variations;
- To maintain and enhance the ORIGEN2 model and code system for calculating detailed isotopic generation and depletion, for application to spent fuel and all categories of radwaste.

In addition to these four tasks, the program provides direct support for special needs for directly derived information and also carries out related studies, under separate funding, that draw heavily on IDB

background information and/or capabilities, but go beyond the scope of the IDB Program itself. Two of these separately funded studies are also described in Chapter 15: TRU Commercial Fuel Cycle Studies and TRU Isotope Decay Chains.

A major project each year that draws heavily on all four of the task areas outlined above is the preparation of an annual report covering all domestic spent fuel and radwaste inventories, projections, and characteristics; the third such report was issued recently.¹ This project uses the capabilities and data developed by these task areas, and brings existing capabilities to a focus. It also serves as an ideal vehicle to present, under a single cover, a huge amount of related data. No other publication in the field has this scope of coverage. The report also helps to identify weak areas, where more or better data are needed, where improved or modified data handling capabilities are required, and it permits (and encourages) a critical review of the data by all interested users, to everyone's benefit. An overall summary table, taken from the latest report, is given in Table 12.1. Many of the characteristics were calculated and projected using newly developed ORGENTRE (Oak Ridge Generation and Treatment) codes. The waste inventory data are acquired through the cooperation of the lead offices that were established by the DOE to oversee the management of the various types of waste. It is expected that future reports containing additional information will be published periodically as the ORGENTRE system becomes fully developed and operational. A special session on Radwaste Inventories and Projections² was held at the ANS National Meeting in Los Angeles, June 1982, including three papers describing this work.³⁻⁵

Table 12.1. Spent fuel and radwaste inventories as of December 31, 1981

<u>Spent fuel</u>	<u>Mass, MTU</u>	<u>Activity, kCi</u>	<u>Heat, kW</u>
BWRs (18,700 assemblies)	3,300	3,360,000	12,500
PWRs (10,900 assemblies)	4,500	6,960,000	26,700
<u>High-level waste</u>	<u>Volume, m³</u>	<u>Activity, kCi</u>	<u>Heat, kW</u>
Savannah River	106,000	982,000	2,900
Idaho CPP	12,000	64,000	210
Hanford	183,000	531,000	1,640
Nuclear Fuel Services	2,190	39,000	115
<u>Transuranic waste</u>	<u>Volume, m³</u>	<u>Activity, kCi</u>	<u>TRU elements, kg</u>
DOE, buried	299,000	400	1,000
DOE, stored	67,000	1,000	910
Commercial, buried	(in with LLW)	---	130
<u>Low-level waste</u>	<u>Volume, m³</u>	<u>Activity, kCi</u>	<u>Land used, ha</u>
DOE sites	1,607,000	10,700	159
West Valley (closed 1975)	66,500	580	3
Maxey Flats (closed 1977)	135,000	2,400	6
Sheffield (closed 1978)	88,000	60	4
Barnwell, SC	363,000	3,000	19
Beatty, NV	93,000	370	11
Richland, WA	102,000	1,020	11
<u>Remedial Action Program</u>	<u>Volume, m³</u>	<u>Activity, kCi</u>	<u>No. of sites</u>
UMTRAP	22,000,000	150	24
FUSRAP	452,000	---	31
SFMP	1,560,000	---	~500
<u>Active mill tailings</u>	<u>Volume, m³</u>	<u>Activity, kCi</u>	<u>Heat, kW</u>
21 active mills	87,000,000	1,000	20

The IDB Program obviously must draw upon data sources outside of its own funding area, and work closely with many other programs and with contractor lead sites and DOE Field Offices and HQ staff in order to achieve effective integration. To facilitate this, a steering committee has been set up, with membership representing all major programs and functional areas directly related to spent fuel and radwastes. The committee is backed up by technical contacts and DOE contacts. As programmatic changes occur, the committee membership also changes to reflect this. The makeup of the committee as of June 1982 is given in Ref. 1.

A series of computer codes and data files called ORGENTRE is being developed to handle this information and to project nuclear waste generation and treatment options. It consists of the following major components:

1. A series of waste generation codes and data bases comprising (a) characteristics of the existing inventory, (b) a nuclear materials projection code to calculate the fissile and fertile materials throughputs of all fuel cycle facilities for any given nuclear power buildup and reactor mix, (c) a defense waste projection code, and (d) an institutional/industrial waste projection code.
2. A waste treatment simulation code to calculate the characteristics of secondary wastes that are generated as a consequence of any proposed treatment process.
3. A system data base composed of several libraries containing fuel cycle facility and waste treatment process flowsheet data, and tables of pertinent nuclear and engineering conversion constants.

4. An economics data file and code to determine the cost of various treatment options.
5. An output file containing the identity and characteristics of all the wastes that have been finally generated and projected, and a data base manager program that is needed to supply tabular and graphical representations of these data.

The overall data flow and information handling scheme is shown in Fig. 12.1. Isotope generation and depletion are handled by the ORIGEN2 code system and data files are managed with SAS (Statistical Analysis System). SAS has been an extremely helpful tool. It can take as input virtually any pre-existing computer data file, generates our own tabular output (which we then print directly on a CPT-8000 word processor through a link to a PDP-10 large computer), can produce graphics (see Ref. 1 for examples of pie charts and line graphs), and will mesh with any future DBMS (Data Base Management System) the program may elect to use.

The ORIGEN2 code for isotope generation and depletion calculations predates this program⁶⁻⁸ but is now supported by the IDB Program because it is an essential element in ORGENTRE. ORIGEN2 is the most widely used code for the calculation of isotopic composition of spent reactor fuel. We also use it to compute decay of the various radioactive wastes. It can also apportion the total spent fuel assemblage into various components, such as fuel, cladding, and structural material or, for the reprocessing case, into recovered U and Pu, HLW, TRU waste, effluents, and structural material. The ORIGEN2 code has been modeled for many

Fig. 12.1. Data Flow for Integrated Data Base Program

reactor systems and variations thereof, including many LWR and LMFBR cases. Four reports have been issued recently dealing with specific cases.⁹⁻¹²

A bibliographic file is also being created, with emphasis on inventories, projections, and characteristics. Abstracts, key words, and other pertinent information are stored in a computer file that is designed to facilitate machine searches. A two-volume bibliography^{13,14} (abstracts and indices) has been published and also an update.¹⁵ The program was also responsible this past year for publication of the Radioactive Waste Management Newsletter.¹⁶

12.3 PUBLICATIONS AND REFERENCES

1. Spent Fuel and Waste Inventories, Projections, and Characteristics, DOE/NE-0017-1 (October 1982).
2. K. J. Notz, Radwaste Inventories and Projections: An Overview, ORNL/TM-8322 (July 1982).
3. E. F. Mastal, "Data Base Needs and Functions; National Planning," ANS National Meeting, Los Angeles, CA, June 1982; Proceedings, Vol. 42, pp. 80-81.
4. K. J. Notz, W. L. Carter, and A. H. Kibbey, "Integrated Data Base for Spent Fuel and Radwaste: Inventories," ANS National Meeting, Los Angeles, CA, June 1982, Proceedings, Vol. 42, pp. 81-83.
5. C. W. Forsberg, C. W. Alexander, and G. W. Morrison, "Integrated Data Base Projections," ANS National Meeting, Los Angeles, CA, June 1982; Proceedings, Vol. 42, pp. 83-84.
6. M. J. Bell, ORIGEN - The ORNL Isotope Generation and Depletion Code, ORNL-4628 (May 1973).

7. A. G. Croff, A User's Manual for the ORIGEN2 Computer Code, ORNL/TM-7175 (July 1980).
8. A. G. Croff, ORIGEN2 - A Revised and Updated Version of the Oak Ridge Isotope Generation and Depletion Code, ORNL-5621 (July 1980).
9. A. G. Croff and C. W. Alexander, Decay Characteristics of Once-Through LWR and LMFBR Spent Fuels, High-Level Wastes, and Fuel Assembly Structural Material Wastes, ORNL/TM-7431 (November 1980).
10. A. G. Croff and M. A. Bjerke, Alternative Fuel Cycle PWR Models for the ORIGEN Computer Code, ORNL/TM-7005 (February 1980).
11. A. G. Croff, J. W. McAdoo, and M. A. Bjerke, LMFBR Models for the ORIGEN2 Computer Code, ORNL/TM-7176 (October 1981).
12. A. G. Croff, M. S. Liberman, and G. W. Morrison, Graphical and Tabular Summaries of Decay Characteristics for Once-Through PWR, LMFBR, and FFTF Fuel Cycle Materials, ORNL/TM-8061 (January 1982).
13. C. A. Johnson and P. A. Garland, Radioactive Waste Management Integrated Data Base: A Bibliography, ORNL/TM-7385/V1 (Indices) (September 1980).
14. C. A. Johnson and P. A. Garland, Radioactive Waste Management Integrated Data Base: A Bibliography, ORNL/TM-7385/V2 (Citations) (September 1980).
15. C. A. Johnson, Radioactive Waste Management: A Bibliography for the Integrated Data Base Program, ORNL/TM-7385/V3 (Update) (October 1981).
16. "Radioactive Waste Technology Program Newsletter," Issue No. 15 (November 1981); Issue No. 16/17 (May 1982); Issue No. 18 (August 1982).

12.4 REFERENCES

None.

13. LOW-LEVEL WASTE MANAGEMENT PROGRAM

L. J. Mezga, N. H. Cutshall

13.1 INTRODUCTION

In FY 1982 two tasks, (1) the Low-Level Waste Information Center and (2) the Tn State Screening project were performed as Low-Level Waste Commercial Nuclear Waste Programs. These tasks were carried out under the overall management of the Program Manager at DOE's Idaho Operations Office, the lead contractor, (EG&G, Idaho) at the Idaho National Engineering Laboratory, and DOE's Oak Ridge Operations Office.

13.2 PROGRESS

Technical progress for each of the tasks of this program for FY 1982 follows:

13.2.1 Low-Level Waste Information Center (C. S. Fore)

The Low-Level Waste Information Center continued in its function to support the U.S. DOE's National Low-Level Waste Management Program (LLWMP). As of January 31, 1982, LLWMP funding for this effort was ceased, and all future tasks were supported through U.S. DOE's Integrated Data Base Program.

The task provided information support to the Low-Level Waste Management Program. This was achieved by maintenance of a bibliographic data base containing scientific literature pertinent to the Low-Level Waste Management Program. The data base makes this information readily available to researchers and decision makers involved with low-level radioactive wastes. Work is divided into three activities: (1) information collection and analysis, (2) publications, and (3) request handling.

The accomplishments for the funded period October 1, 1981 to February 1, 1982 were as follows: (1) identified, collected, analyzed, abstracted, indexed, and disseminated technical information relevant to low-level radioactive waste technology - approximately 200 new documents were received and 150 abstracts were written and entered into the online data base; (2) compiled and published one newsletter on radioactive waste technology receiving a distribution of approximately 1000; and (3) responded to 300 information requests for governmental, academic, industrial, and foreign affiliations.

13.2.2 State Screening Project (J. L. Cannon, W. P. Staub, P. R. Intemann, and L. E. Stratton)

This project supports the DOE Low-Level Waste Management Program responsibility for providing basic technical assistance to requesting states and regional groups considering sites for disposal of low-level radioactive waste. This support to states is the responsibility of EG&G-Idaho. The ORNL team works with EG&G staff in this project. The technical assistance includes: (1) the development of screening parameters and a screening process, based on 10 CFR 61 and work in the low-level waste management program, applicable to the existing geographic, hydrologic and geologic systems of the particular state and/or region; and (2) application of the screening process through a reconnaissance-level investigation to identify potentially suitable areas and sites for low-level waste disposal. The current focus of the project is the State of Tennessee which is used as a model to demonstrate a procedure for locating low-level waste disposal sites. The results of the project are intended to be of direct use to the State of Tennessee, and will be available to other states as general information and support for site selection projects.

A formal site selection process for a low-level waste disposal facility may be formulated as three levels of consideration: (1) the region is screened for suitable areas; (2) the areas are screened for potential viable sites on a more detailed scale; and (3) the potential viable sites are screened to yield recommended sites. As shown in Fig. 13.1, each step progresses to a smaller geographic unit and involves more detailed information. Table 13.1 lists the criterion measures which are to be used at each screening level of this process. The criterion measures were developed from consideration of the Nuclear Regulatory Commission's proposed waste-disposal regulation 10 CFR 61.

The first level of screening in this site selection process was conducted for the State of Tennessee to ascertain the subregions within the state most likely to have suitable areas. The subregions were defined as physiographic provinces which are finite divisions of land with distinct geologic and geomorphic characteristics (Fig. 13.1). Based on the screening criteria and screening factors (Table 13.1), only the eastern Mississippi Embayment was found to have all of the desirable natural features to meet the criteria for shallow land burial of low-level waste. The remaining physiographic provinces were excluded on the basis of one or a combination of the following: (1) geological hazards, (2) complex features such as karst topography and faults that may not be readily amenable to reliable hydrogeologic analysis, and (3) insufficient overburden. Figure 13.2 shows the composite suitability map.

Figure 13.1. Screening levels and analysis.

Table 13.1 Criteria measurements: information readily available
for initial screening and information required for site licensing

Screening level	IA	IB	II	III	Primary Site Characterization
Criterion Information	Available for Screening			Required for Licensing	
1. Volume (including buffer zone)					
Areal extent			+	+	+
Soil thickness	+	+	+	+	+
2. Depth to water table		+	+	+	+
3. Flood hazard					
Drainage divides		+		+	
100-yr flood plain		+	+	+	
4. Erosion potential		+	+	+	+
5. Geohydrologic predictability				+	+
Solution cavities			+	+	+
General geology	+	+	+	+	+
Faults and fractures	+	+	+	+	+
6. Geologic hazards					
Seismicity	+			+	
liquefaction	+			+	
Slope instability	+			+	
Subsidence	+			+	
karst topography	+			+	
groundwater withdrawal	+			+	
mines	+			+	
7. Geochemical Retardation				+	+
8. Population					
Present	+			+	+
Trends		+		+	
9. Land Use and Resource Development	+	+		+	+
10. Transportation	+				
11. Laws					
Federal	+				
State	+				
Local				+	
12. Regulations					
Federal	+				
State	+				
Local				+	

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HIGH SUITABILITY

LOW SUITABILITY

INTERMEDIATE SUITABILITY

Figure 13.2. Composite suitability map for Tennessee.

14. TRANSPORTATION PROGRAM

14.1 INTRODUCTION

The goal for this task of the Transportation Technology Center (TTC) transportation program is to evaluate the movement of commercial nuclear waste from point of generation to final isolation, utilizing the various computer codes already developed for these studies; where existing codes are not adequate to generate the required information, they will be modified appropriately.

Major objectives are to complete the logistics support studies as required, upgrade and utilize the full-scale repository logistics model, obtain up-to-date shipping cost data, apply the SCOPE (Shipping Cask Optimization and Parametric Evaluation) cask design code to current needs, and apply the TEF computer models to current needs.

14.2 PROGRESS

Technical progress for each of the tasks of this program for the fiscal year just ended follows.

14.2.1 Commercial Waste Transportation Studies (L. B. Shappert, D. S. Joy, J. F. Walker, S. M. Gibson, and P. E. Johnson)

In the transportation/logistics area, the major task in FY 1982 was the Test and Evaluation Facility (TEF) Logistics Study. A computer program for simulating the flow of packages from waste sources to packaging facilities and TEF sites was developed and made operational three months ahead of schedule. Over 200 different TEF runs, each simulating a different transportation scenario, were completed by February 1982 to assist ONWI in evaluating the EMAD facility as a possible waste packaging site. Additional analyses were performed evaluating six hot cells located around the United States as possible

packaging facilities in the TEF transportation system. A large number of conceptual cask designs were evaluated with the TEF model to assist TTC and General Atomic in preparing the Cask Acquisitions Report. Several alternate cask designs were included for each of the proposed waste packages. Several modes of transportation, including legal and overweight truck and rail casks, were included in the study.

Approximately 60 special maps were prepared showing the highway network in major cities around the country. These maps also included population densities generated from 1980 Bureau of the Census data. This information was used by both TTC and the Department of Transportation in formulating responses to questions concerning the proposed DOT regulation, HM 164.

The NUS waste management model RAWSYM was successfully installed on the ORNL computer system. A number of test cases were run and checked with the data published by NUS. The results indicated that the ORNL version was identical to the program being operated by NUS.

The RAWSYM model is being modified to meet the specific needs for repository transportation scenarios funded by the TTC. The first phase of this modification was completed in FY 1982 and included the incorporation of the latest DOE spent fuel data base. The projected spent fuel shipping schedules match the estimated spent fuel storage experiments (14.4.1).

A number of preliminary transportation calculations were performed in support of the ONWI salt repository site recommendation study. Information generated included transport costs, distance traveled, and cask fleet requirements. Seven different salt sites as well as the basalt and tuff sites were included in this analysis.

The outcrop formations of the remaining physiographic province, the eastern Mississippi Embayment, were then examined in a more detailed manner. The process identified the Coon Creek Formation outcrop band as the best candidate area for further screening for low-level waste sites. This area is remote from population centers; major transportation systems are available; the risk of flooding is minimal; overburden thickness is adequate; erosion resistance is high; the water table is deep; the area is well drained; a relatively impermeable stratum lies above the regional aquifer offering protection from contamination; there are no apparent geologic hazards; and reliable groundwater monitoring and predictability are achievable.

13.3 PUBLICATIONS

None.

13.4 REFERENCES

None.

A computerized technique to evaluate the distribution of population densities along a transportation route were developed. Approximately 54 routes were generated to check the population density assumptions used in the RADTRAN program at Sandia National Laboratory. A special set of routes was also prepared showing the population density distribution along both highway and rail routes which would likely be used in transporting waste to a repository site.

14.3 PUBLICATIONS

1. J. W. Cashwell and D. S. Joy, "Impacts of Nuclear Materials Flow on Transportation Routing Alternatives," Waste Management '82, Vol. 1, p. 389.

14.4 REFERENCES

1. Spent Fuel Storage Requirements, DOE/RL-82-1 (June 1982).

15. TRU WASTE MANAGEMENT PROGRAM

W. W. Pitt

15.1 INTRODUCTION

The Commercial TRU Waste Management Program for FY 1982 consisted of three tasks:

1. FUECAP Concrete Immobilization - jointly sponsored by the Defense TRU Waste Management Program and reported in Section 10.
2. TRU Commercial Fuel Cycle Studies - provide quantitative data on TRU waste derived from LWR reprocessing and LMFFBR fabrication and reprocessing.
3. TRU Isotope Decay Chains - calculate the detailed alpha decay behavior of 38 isotopes and present results graphically.

15.2 PROGRESS

The progress for each of the tasks in this program for FY 1982 follows:

15.2.1 FUECAP Concrete Immobilization (L. R. Dole, G. C. Rogers, J. H. Kessler, and T. M. Gilliam)

This program was jointly funded by the Defense TRU Waste Management Program and progress is reported in Section 10.2.1.

15.2.2 TRU Commercial Fuel Cycle Studies (K. J. Notz, W. L. Carter and P. J. Johnson)

The objectives of this program for FY 1982 were: (1) to provide quantitative data on TRU waste derived from LWR reprocessing, based on the AGNS (Barnwell) flowsheet, for realistic spent fuel burnups and ages prior to reprocessing and for realistic fuel cycle scenarios; and (2) to

provide qualitative data on TRU waste derived from LMFBR fabrication and reprocessing, with attention directed toward identification of generic differences between the LMFBR and LWR cases. Both objectives used projection scenarios out to year 2020, and these were based on the Integrated Data Base (IDB) treatment (see Section 12).

Detailed flowsheets were prepared for LWR reprocessing, based on the AGNS flowsheet,¹ which is sized for 1500 MTHM/y. All waste streams were quantified (not just the TRU waste streams) in order to achieve an overall material balance. The identities of individual waste streams were retained in order to facilitate any future changes or modifications to the processing flows, separation factors, or distribution coefficients. The overall plant was broken down into six major operations: head end, solvent extraction, waste treatment, off-gas treatment, uranium product recovery, and plutonium product recovery. An overall summary of the waste volumes is shown in Fig. 15.1.

The elemental and isotopic contents of the waste streams were then calculated, assuming spent fuel 16 y out of reactor with an average burnup of 25,200 Mwd/MTHM, made up of two-thirds PWR fuel and one-third BWR fuel. The incoming compositions were calculated using the ORIGEN2 code² and the distributions as a result of reprocessing were also handled using ORIGEN2. The basis for total quantities was the reactor growth projection used by the Integrated Data Base Program³ and the reprocessing plant schedule assumed by that program:

1500 MTHM/y in 1989

1500 MTHM/y additional in 2000

1500 MTHM/y additional in 2004

1500 MTHM/y additional in 2008

WASTE STREAMS

Fig. 15.1. Waste Streams from LWR Reprocessing

For the LMFBR fuel reprocessing case, a qualitative analysis was made to identify generic differences relative to LWRs that will derive from the different fuel (i.e., mixed U, Pu oxide), harder neutron spectrum, and different cladding (sodium-bonded stainless steel). These results and the LWR data will be published in a topical report (see under 15.3).

15.2.3 TRU Isotope Decay Chains (K. J. Notz and C. W. Alexander)

This was a special, short-term task to provide detailed decay behavior of all TRU (or potential TRU) isotopes. The calculations were done via ORIGEN2 for 38 isotopes. The detailed results were summarized on hard copy (a 6-cm-high stack of computer printout for the summary) and then presented graphically, one page per isotope, showing the total alpha for each isotope, as it decays, including daughters. The time frame covered is from 0.1 year to 10^6 years. Each isotope was "started out" at 100 nCi initial activity, either alpha or beta, as appropriate.

The 38 initial nuclides included are listed in Table 15.1. An additional 21 nuclides appeared in the calculations as daughter products, Table 15.2. Two plutonium isotopes are shown in Figs. 15.2 and 15.3, illustrating several features observed in these plots; e.g., slow decay of a long-lived parent, in-growth of both short-lived and long-lived daughters, low initial alpha activity from a parent that first undergoes a beta decay, and near-term growth of total alpha activity. This work will be published as a program document (see under 15.3).

Table 15.1. Initial Nuclides Used in this Study

Ra-226	U-232	Pu-236	Cm-242
Ra-228	U-233	Pu-238	Cm-243
Ac-227	U-234	Pu-239	Cm-244
Th-228	U-235	Pu-240	Cm-245
Th-229	U-236	Pu-241	Cm-246
Th-230	U-238	Pu-242	Cm-247
Th-232	Np-236	Pu-244	Cm-248
Pa-231	Np-237	Am-241	Cf-249
Pa-233		Am-242	Cf-250
		Am-243	Cf-251
			Cf-252

Table 15.2. Additional Daughter Nuclides Used in this Study

Bi-211	Po-210	At-217	Ra-223
Bi-212	Po-211	Rn-219	Ra-224
Bi-213	Po-212	Rn-220	Ac-225
Bi-214	Po-213	Rn-222	Th-227
	Po-214	Fr-221	
	Po-215		
	Po-216		
	Po-218		

ORNL DWG 82-14532

DECAY OF 100 NANOCURIES OF PU-238

Fig. 15.2. Decay of 100 Nanocuries of Pu-238

ORNL DWG 82-14535

DECAY OF 100 NANOCURIES OF PU-241

Fig. 15.3. Decay of 100 Nanocuries of Pu-241

16. WASTE ISOLATION PROGRAM

S. H. Stow

16.1 INTRODUCTION

The Waste Isolation Program is supported through the Office of Nuclear Waste Isolation (ONWI) at Battelle Memorial Institute in Columbus, Ohio. ONWI's efforts are directed toward the siting of a geologic repository for commercial nuclear waste. ORNL supports this endeavor with a wide variety of tasks.

16.2 PROGRESS

Technical progress for each of the tasks of this program for the fiscal year just ended follows.

16.2.1 Regional Environmental Characterization Studies - Nuclear Waste (J. R. Krummel and S. G. Hildebrand)

The objective of this task is to characterize the environment of the Southern Piedmont region that includes a number of granitic formations which have geologic characteristics potentially suitable for permanent disposal of nuclear wastes. The region will be screened using environmental and geologic criteria in order to identify the most favorable areas for further study. This task supports the ONWI overall goal of providing a significant degree of assurance that high-level nuclear wastes can be isolated from the biosphere in a safe and environmentally acceptable manner.

Information contained in the regional environmental characterization report was used to screen geologic formations which have potential as high-level nuclear waste repositories to select formations suitable for further detailed analyses in the area phase of this study. The selection criteria and the screening process was included in the summary and recommendation report.

16.2.2 Institutional Arrangements for Nuclear Waste Siting
(S. A. Carnes and M. Schweitzer)

The National Waste Terminal Storage Program recognizes the need to reach consensus not only on the technical issues but on institutional and social issues as well. Our design and testing of possible incentive arrangements for siting hazardous waste facilities must, therefore, include an integration of economic, institutional, and public participation factors into each alternative considered. The objective is the development of a range of evaluated options for inclusion in the consultation and concurrence and other siting processes.

There are five important findings in this research: (1) the considerable uncertainty that remains in the repository design and siting process indicates a need for continued research to lessen uncertainty, and flexibility in job training program design as uncertainty is reduced; (2) there are likely to be multiple major factors that need to be involved in job training program design (i.e., the Department of Energy, repository state and local governments and their training institutions, labor unions, and the prime contractor and its subcontractors); (3) mechanisms for resolving potential

conflicts among these major actors need to be developed; (4) program design and implementation decisions need to be documented carefully to protect each program participant's interest; and (5) given the number and diversity of major actors, a clear assignment of responsibilities for program design and implementation must be made.

16.2.3 Evaluation of Avery Island Coupons (J. C. Griess)

The objective is to obtain the necessary corrosion data so that various multibarrier waste package emplacement concepts in domal salt can be evaluated.

The tests showed that only carbon steel experienced appreciable attack; maximum penetration during the three-year test was only 200 microns. All other test materials (titanium, Zircaloy-2, Hastelloy C-276, E-Brite 26-1, and T-304L stainless steel) showed no change, except a few isolated areas of attack were noted on E-Brite 26-1 and T-304L stainless steel, none of which exceeded 25 microns in depth.

16.2.4 Effects of Water in Salt Repositories (C. F. Baes, Jr.)

It is the objective of this program to predict from experimental and modelling studies those consequences of the presence of water in a salt repository that might compromise the integrity of the repository.

While the generalization of the results is not yet complete, it seems clear that the consolidation of polycrystalline salt down to void fractions of approximately 0.05 can be represented adequately by the expression already presented. Since in a number of tests

consolidation has been carried almost to completion with continuous expulsion of brine, it appears that the incorporation of brine within crystals is not an important trapping mechanism. It should be possible to approximate the permeability of consolidated salt as a function of applied stress, time, and temperature in terms of the expression found for the permeability of crystal interfaces. This permeability probably will determine the final rate of consolidation.

16.2.5 Salt Model Pillar Studies (T. F. Lomenick)

The objective of this work is to provide data for the proper design of excavated cavities in rock salt repositories that are candidates for the storage/disposal of radioactive wastes.

Recently a careful examination of the creep records of the long-term tests revealed the existence of episodes of transient creep occurring between periods of steady behavior. The potential importance of these transient creep episodes in repository design and in the verification of computer codes to be used in design justified a more in-depth investigation. Such episodic creep had not previously been reported for salt and consequently a question arose as to whether transient episodes reflected material behavior or some aspect of the test equipment such as sticking and slipping of the load ram or environmental parameters such as temperature or humidity fluctuations in the laboratory.

In order to investigate episodic creep more completely, it was desirable to automate the data acquisition and control the tests to achieve more accuracy and better control.

The most suitable system would include a data acquisition and control system with printing, plotting, and magnetic storage capability controlled by a dedicated micro computer capable of performing data analysis. By using as much software control as possible, such a system provides a great amount of flexibility for possible future data acquisition and control needs.

16.2.6 Expected Repository Environments for Commercial Wastes, Savannah River, and Other Wastes (H. C. Claiborne)

The primary objective is to provide information in designing experiments and repositories and in licensing procedures based on calculated expected repository environments for the disposal of both commercial and defense high-level wastes and spent fuel in salt formations. The predicted environments or effects include near- and far-field temperature histories, brine chemistry, brine migration rates, fluid pressure histories, and nuclear radiation fields. A series of reports has been issued describing the environments.

16.2.7 Technical Topical Report on Brine Migration (H. C. Claiborne and G. H. Jenks)

This project was initiated to provide technical data on brine migration in a salt repository as input to a licensing topical report. A report was issued describing the technical findings.

16.2.8 System Model to Assess Flooding in a Waste Repository (H. C. Claiborne)

The objective of this project is to develop the methodology and appropriate computer codes for analyzing the effects of freshwater intrusion into a salt repository for nuclear waste that also has applicability for other geologic media. The methodology to be developed will include the effects of mechanical responses to the mine, the analysis of potential flow patterns in both open rooms and rooms backfilled with crushed salt or other materials and the potential radionuclide concentrations that could result from the assumed flooding scenarios and waste package leaching models.

Salt dissolution mechanisms were examined, and the sequence of solutioning was divided into two phases--essentially an isothermal phase during flooding and after filling and a heatup phase causing additional dissolution of salt. Models for the two phases were identified. The literature on leaching of radionuclides was examined and appropriate models were identified. Evaluation of the experimental data on the permeability of crushed salt as a function of time while being stressed was completed. Considerable progress was made on developing a computer code for simulating the flooding process. This code (REFS) will be based on the existing code HYTAQ.

16.2.9 Sensitivity and Uncertainty Analysis Methods Development (E. M. Oblow)

The objective of this task is to develop the GRESS computer language to automatically calculate the sensitivities of waste

repository model performance with respect to model data. Results of this task will be central to the development of a comprehensive methodology for sensitivity and uncertainty analysis for waste repository design licensing. The development of the GRESS language will automate the application of adjoint sensitivity theory to large scale risk analysis problems. It will allow sensitivities to be produced for all model data thereby focusing licensing analysis on the most important parameters and their uncertainties. Developments in this task will make adjoint sensitivity theory competitive with statistical uncertainty methods in applicability to large scale models. The inherent advantage of adjoint methods in study completeness will then allow a comprehensive sensitivity and uncertainty analyses to be performed on a waste disposal design for licensing.

This task began in June 1981 with the development of a demonstration version of the GRESS automated language approach to adjoint sensitivity analysis. The program was halted September 30, 1981 with the completion of GRESS Version A. After an evaluation of adjoint sensitivity methods by ONWI in April 1982 plans were made for full scale development and implementation of this approach for repository design licensing. This overall program will begin October 1, 1982. The small task described in this report was begun in the last quarter of FY 1982 to evaluate the feasibility of demonstrating a fully developed version of the GRESS language on the large scale hydrology modeling code SWENT.

16.2 PUBLICATIONS

D. J. Bjornstad and K. Johnson, "Payments in Lieu of Taxes Between Federal and Local Governments in the Siting of Federally-Owned Waste Isolation Facilities," Review of Regional Studies, v. 12, 1982.

D. J. Bjornstad and G. Goss, "Issues in the Use of Payments in Lieu of Taxes to Provide Nuclear Waste Facility Siting Incentives," Radioactive Waste Management, December 1981.

B. Beyerlein and H. C. Claiborne, "The Possibility of Multiple Temperature Maxima in Geologic Repositories for Spent Fuel from Nuclear Reactors," Jour. of Heat Transfer, February 1982.

S. A. Carnes, et al., Incentives and the Siting of Radioactive Waste Respositories, ORNL-5880.

S. A. Carnes, et al., "Framework for Evaluating the Utility of Incentive Systems for Radioactive Waste Repository Siting," Waste Management, 1982, Vol. 2, Roy G. Post, Ed. (Tucson: University of Arizona, 1981)

S. A. Carnes et al., "Incentives and Nuclear Waste Siting: Prospects and Constraints," submitted to Energy Systems and Policy 16, July 1982.

H. C. Claiborne, The Efficacy of Backfilling and Other Engineered Barriers in a Radioactive Waste Repository in Salt, ORNL/TM 83/2.

E. D. Copenhaver et al., "Framework for Evaluating the Utility of Incentive Systems for Radioactive Waste Repository Siting," In Advances in the Science and Technology of the Management of High-Level Nuclear Waste, Peter L. Hofman, ed. (Columbus, Ohio: Office of Nuclear Waste Isolation, in press).

15.3 PUBLICATIONS

W. L. Carter, P. J. Johnson, and R. H. Rainey, Quantities and Characteristics of Transuranic Wastes from Commercial Fuel Cycles, ORNL/TM-8580 (to be published).

C. W. Alexander and K. J. Notz, Alpha Decay of 38 Nuclides, ORNL/NFW-82/32 (to be published).

15.4 REFERENCES

1. W. H. Carr, Estimation of the Nuclear Waste Types, Characteristics, and Quantities from the Barnwell Nuclear Fuel Plant, E 512-09600R (May 1982).
2. A. G. Croff, A User's Manual for the ORIGEN2 Computer Code, ORNL/TM-7175 (July 1980).
3. Spent Fuel and Radioactive Waste Inventories, Projections, and Characteristics, DOE/NE-0017-1 (October 1982).

E. D. Copenhaver et al., "Framework for Evaluating the Utility of Incentive Systems for Radioactive Waste Repository Siting," NWTs Program, DOE/NWTs-15.

L. O. Gilpatrick, C. F. Baes, A. J. Shor, and D. Canonico, The Permeability of Salt Crystal Interfaces to Brine, ORNL-5874.

L. O. Gilpatrick, A. J. Shor, C. F. Baes and D. Canonico, Initial Report on the Permeability of Salt Grain Boundaries, ORNL/NFW-81/32.

J. C. Griess, Evaluation of Corrosion Damage to Materials After Three Years in the Avery Island Salt Mine, ORNL/TM-8351.

G. H. Jenks and H. C. Claiborne, Brine Migration in Salt and Its Implications in the Geologic Disposal of Nuclear Waste, ORNL-5818.

J. R. Krummel, M. Nungesser, C. Hunsacker, and C. Tucker, Regional Characterization of the Southern Appalachian Crystalline Region, ONWI-343.

T. F. Lomenick and J. Russell, Measurement of Long Term Creep in Salt Pillars, NWTs Program, DOE/NWTs-15.

L. Rickertsen, L. Arbutal, and H. C. Claiborne, Expected Near-Field Thermal Environments in a Sequentially Loaded Nuclear Waste Repository in Salt, ORNL/TM-8083.

L. Rickertsen, O. Misplon, and H. C. Claiborne, Expected Very-Near-Field Thermal Environments in a Salt Repository for Advanced Spent Fuel and Defense High-Level Waste Package, ORNL/TM-8109.

16.4 REFERENCES

None.

17. SURPLUS FACILITIES MANAGEMENT PROGRAM

J. H. Coobs

17.1 INTRODUCTION

This is the second of two programs that are concerned with the management of surplus facilities. The facilities in this program are those related to commercial activities, which include the three surplus experimental and test reactors [(MSRE, HRE-2, and the Low Intensity Test Reactor (LITR))] and seven experimental loops at the ORR. The program is an integral part of the Surplus Facilities Management Program, which is a national program administered for DOE by the Richland Operations Office. Objectives of the program are to provide for surveillance and maintenance of surplus radioactively contaminated DOE facilities awaiting decommissioning in order to assure safe conditions and to implement a structured decommissioning program to accomplish disposition of all surplus DOE-owned facilities. The decommissioning operations are supported by financial estimating, planning and scheduling, engineering, and technology development activities.

17.2 PROGRESS

Technical progress for the task of this program for fiscal year 1982 follows.

17.2.1 Commercial Surplus Facilities Surveillance (R. V. McCord and L. P. Pugh)

This task includes survey and inspection of surplus facilities on a regular basis, collection and analysis of samples from waste streams,

replacement of exhaust air filters on a regular basis, and maintenance of facilities as indicated by procedures and results of inspections.

Routine surveillance and maintenance activities were continued at several facilities. The annual fuel salt reheat and containment leak test was completed at the MSRE. A leaking shield plug at the ORR north beam hole HN-1 was replaced with a new plug. This facility had previously been used for the molten salt loop experiment. Excess components and exposed piping at the LITR were removed and disposed of as solid waste. A radiological survey of the HRE facilities was completed except for retrieval and analysis of subsurface soil cores.

17.3 PUBLICATIONS

None.

17.4 REFERENCES

None.

18. URANIUM MILL TAILINGS PROGRAM

J. H. Coobs

18.1 INTRODUCTION

The Uranium Mill Tailings Remedial Action Program (UMTRAP) is a national program dedicated to the cleanup and/or stabilization of 25 abandoned mill tailings sites in ten states. The program is administered for DOE by the Albuquerque Operations Office. Program participation by ORNL during FY 1982 consisted of two small tasks: one on evaluation of liners for tailings piles and a second on information support for all aspects of the program. The liner technology task was intended to address the long- and short-term changes in properties of liner materials, but the scope of work was limited to an evaluation and assessment of grouted in-situ liners as a remedial action alternative. The second task provides information support to the program by maintaining a computerized data base, conducting searches of the data base as a service to users, preparing an annotated bibliography covering additions to the data base, and providing technical support at conferences.

18.2 PROGRESS

Technical progress for each of the tasks for this program for fiscal year 1982 follows.

18.2.1 Liner Evaluations for Uranium Mill Tailings (T. Tamura and W. J. Boegly, Jr.)

This task was responsive to a request from the UMTRA Program Office to prepare an assessment of the possible utilization of grouted in-situ

liners as a remedial action alternative. The work was carried out by collecting and evaluating information on grouting technology obtained through literature searches and from discussions with experimentalists and practitioners of grouting. This information and an assessment thereof were documented in a draft report that was submitted for review and will be published early in FY 1983. The report discusses grouting equipment and procedures and the characteristics of grouts and their compatibility with mill tailings piles, and presents an economic assessment of grouting as a remedial action and recommendations for implementation of the method.

18.2.2 UMTRAP Information Support (P. T. Owen)

The Remedial Action Program Information Center (RAPIC) was supported jointly during FY 1982 by UMTRAP, by the Surplus Facilities Management Program, and by the Formerly Utilized Sites Remedial Action Program, thus enabling the center to perform a full spectrum of information support activities. The data base was expanded by abstracting and indexing 1037 documents, and 502 information requests were answered. Abstracting continued until June when preparations for the publication of volume three of the bibliography were begun. The bibliography was published and 971 copies were distributed. A revised distribution list for RAPIC publications is being maintained as a separate data base.

Information support at conferences consisted of attendance by two representatives of RAPIC at the International Symposium on Management of Waste from Uranium Mining and Milling, held May 10-14, 1982, in

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Albuquerque, New Mexico. RAPIC staff manned an information booth, which included a display of RAPIC and UMTRAP literature, a poster exhibit on RAPIC and UMTRAP activities, and video tapes for individual viewing on uranium mill tailings and decommissioning technologies. Computerized literature searches and copies of RAPIC publications were provided to conference attendees upon request.

18.3 PUBLICATIONS

T. Tamura and W. J. Boegly, Jr., In-Situ Grouting of Uranium Mill Tailings Piles - An Assessment, ORNL/TM-8539 (in publication).

P. T. Owen et al., Nuclear Facility Decommissioning and Site Remedial Actions: A Selected Bibliography, Vol. 3, ORNL/EIS-154/V3 (September 1982).

18.4 REFERENCES

None.

19. FORMERLY UTILIZED SITES REMEDIAL ACTIONS PROGRAM

19.1 INTRODUCTION

Low-level radioactive contaminants associated with past activities at the Middlesex Sampling Plant Site are currently stored on a specially designed and constructed asphalt pad at the south end of the facility. Although this means of containment provides an interim solution to the problem, it is apparent that plans for the ultimate disposition of the wastes must be developed. One of the most promising alternatives for the permanent disposal of these wastes as well as FUSRAP wastes at other sites is in situ stabilization. This report summarizes data on the critical elements of geology and hydrology at the Middlesex site and outlines a plan for their possible emplacement.

19.2 PROGRESS

Technical progress for each of the tasks of this program for this fiscal year just ended follows.

19.2.1 Stabilization Studies at Middlesex, New Jersey Sampling Plant

The former Middlesex Sampling Plant Site (MSPS) occupies a 9.6-acre tract adjacent to the New Jersey Central Railroad and Mountain Avenue within the Borough of Middlesex, New Jersey. The facility was formerly used in the 1940's and 1950's for the processing and storage of uranium and thorium ores, but is currently utilized as a site for the storage of low-level radioactive residues that resulted from the cleanup of surrounding properties. The U.S. Department of Energy owns the facility which is one of the first sites to be investigated and to undergo remedial-cleanup measures under the

Formerly Utilized MED/AEC Sites Remedial Action Program (FUSRAP). National Lead of Ohio, Inc. served as the contractor responsible for the two-phase cleanup operations during 1980-81; since then, management responsibilities for the site have been transferred to Bechtel National, Inc. as the Program Management Contractor for FUSRAP.

The general Middlesex-Piscataway area of New Jersey is situated within a geologic-physiographic region known as the Triassic-Newark Basin. It is one of a number of similar elongate basins which are oriented northeast-southwest and extend from Nova Scotia on the north to northern Florida on the south. These basins are developed in both the crystalline-rock (igneous and metamorphic) terrane of the Appalachian Mountains belt (Piedmont) and beneath younger sedimentary units of the Atlantic Coastal Plain. Most of these basins are known to be bounded by steep faults on at least one side and filled with non-marine sedimentary strata of Triassic to Jurassic age. Many of the rock units are red; associated volcanic lava (i.e., Watchung basalts) and shallow intrusives (i.e., Palisades diabase) are found in many areas.

Northeastern New Jersey was also glaciated during the major glacial age, or Pleistocene. As a result, irregular, linear deposits of glacially-derived and glacially-transported sediment are found to the north of the Middlesex-Piscataway area. These so-called morainic deposits also provided a source of sediments which glacial meltwater redeposited out in front of the moraines. These rewashed, water-lain segments are termed outwash deposits and they typically overlie the red-colored bedrock of the Newark Basin in this area.

Soils found throughout this part of New Jersey are derived from the biochemical weathering of both the Triassic bedrock and these unconsolidated glacial-outwash sediments. In addition to the thin veneer of glacial outwash and soils (in the agronomic sense), man's activities have added backfill (soil in the engineering sense) and other construction materials (crushed stone or aggregate, cinders, asphalt, etc.) in certain localized areas.

The MSPS proper is underlain by a thin covering of artificially-added construction materials, soil, and silty glacial sediments, all of which overlie at shallow depth a bedrock unit known as the Brunswick Shale. As indicated by the boreholes drilled during past studies, the depth to this red-colored bedrock unit varied from 3.0 to 7.5 feet throughout the MSPS.

Based upon the borehole data collected onsite and regional information about the Brunswick Shale in adjacent areas, red shale (or mudstone) is the principal rock type in this geologic formation. Although sandstones and siltstones are found in this formation elsewhere, neither of these lithologies were encountered in any of the five deep boreholes at the MSPS. Thus, information collected to date indicates that red shale is the only rock type present within the shallow (<50 feet) bedrock at the site.

Immediately overlying this red shale is a thin (<1 foot) zone of weathered, disaggregated shale. Where the veneer of glacial sediments is absent, this zone of weathered shale, especially at its more shallow positions, appears as a red clayey silt (or silty clay) containing fragments of red shale. Under the current Soil

Conservation Service designation, this thin, shale-derived soil is called the Reaville Series.

Where not fractured or jointed, the shale of the Brunswick Shale is deep red, compact, silty, and impermeable. Cores recovered from the deep boreholes onsite did, however, reveal zones of green coloration, white calcareous coatings (suggestive of secondary mineralization filling joint openings), and black carbonaceous films as departures from the description above. Another exception was provided by borehole DOE-6A, which apparently penetrated a 9-foot zone of fractured-jointed shale. Hydraulic testing confirmed the presence of this porous-permeable interval.

Discovery of a zone of fractured-jointed shale in the bedrock at MSPS was not unexpected. To the north and northwest, the Brunswick Shale exhibits joints in outcrops and is also interrupted by the three Watchung Mountain lava flows (Van Houten, 1969). Elsewhere, the Brunswick Shale is an aquifer from which potable ground water is obtained; the occurrence and movement of this ground water is clearly associated with the presence of joints within an otherwise permeable sequence of fine-grained rock types (Vecchioli, Carswell, and Kasabach, 1969). A similar relationship between ground-water occurrence and secondary porosity within the Brunswick Shale as caused by jointing has been identified in Essex County, New Jersey to the northeast of the Middlesex-Piscataway area (Nichols, 1968). Roy F. Weston, Inc. (1980) also indicated that ground water is obtained from fracture porosity within the Brunswick Shale in western Middlesex County. Several water wells within a two-mile radius of the MSPS are

believed to produce ground water from fractures-joints in the Brunswick Shale (Ford, Bacon, and Davis, Utah, Inc., 1979). In fact, a greater density of boreholes drilled into this bedrock unit at or near the MSPS would probably indicate the presence of additional zones of fracturing and jointing.

Furthermore, the Brunswick Shale is gently inclined within this part of New Jersey. The unit as a whole is tilted so that it dips toward the northwest.

In an early soil survey (Holman and Junikis, 1953), portions of the tract occupied by the MSPS are mapped as having soil derived from thin glacial-outwash sediments that overlie the red Brunswick Shale. In several of the boreholes (Doe-2, 3, 3-A, 4, 4-A, 6, 6-A, and 8) drilled by Roy F. Weston, Inc., a thin (1/2 - 2-1/2 feet) zone of gray clay-rich silt is present and may well represent this glacial parent material and/or soils. Given the amount of past construction activities, including grading, backfilling, and asphalt paving, it is difficult to clearly establish the above relationship.

By utilizing the 20 boreholes drilled at the MSPS, weekly water-level measurements were taken during June and July of 1980 and compared against precipitation records from the nearby Newark International Airport (Roy F. Weston, Inc., 1980). Several slug tests were also performed in selected boreholes. These data are the principal means by which the nature of the ground-water hydrology at the MSPS has been established.

In summary, the geologic and hydrologic setting of the MSPS is rather straightforward. The site exhibits a rather gentle topography developed upon a thin veneer of soils, artificial fill material, and

weathered shale which collectively overlies a shale-dominated bedrock formation called the Brunswick Shale. the latter is present throughout the site at shallow depth, contains at least one and possibly more zones of higher permeability (or transmissivity if viewed from a hydrologic viewpoint) caused by jointing, and dips toward the northwest. Data collected prior to remedial-action measures and on-site construction indicated the presence of a shallow ground-water system within the overlying unconsolidated materials and a deeper system within the bedrock. Some modification to the entire hydrologic regime at the MSPS has resulted from the several construction activities associated with installation of a disposal pad.

A generalized plan for an in situ stabilization program might include these elements:

1. Excavation outside of the southern portion of the area to a depth of a few feet below grade over several acres to remove any on-site contaminated soil and to acquire adequate natural soil (uncontaminated) for admixing.
2. Backfilling this initial excavation with a 1- to 2-ft layer of additive material (natural and/or artificial) which has been demonstrated to have a high level of ion-exchange capacity with particular regard to the radioactive species within the wastes.
3. Spreading next a thin (6-12 in.) layer of natural soil material over the sorptive buffer layer, followed by alternating layers of waste admixed with the natural soil (and possibly various additives) and soil admixed only

additives. In other words, the waste materials would be contained in a thin-layer stratification sequence.

4. Lateral extension of this approach would be made in order to accommodate all the waste materials currently stored on the pad in the southern part of the area and the contaminated soil under that area. If necessary, a part or all of the southern portion of the area could be utilized so that the resulting layered, in situ stabilized sequence produced as low a topographic increase as possible.
5. Contouring, drainage diversion structures, and other runoff control measures would be incorporated to insure that surface drainage presented a minimal concern with regard to the in situ waste system.
6. A cover of 1 to 2 foot of compacted clay would be added over the upper part of the in situ sequence to significantly lessen the amount of water available for infiltration.
7. Addition with appropriate grading, fertilization, seeding, and mulching of a vegetative cover (grass or other shallow-root plants) would constitute the final step in completing this in situ stabilization approach.

19.3 PUBLICATIONS

T. F. Lomenick, In Situ Stabilization at the Former Sampling Plant Site, Middlesex, New Jersey (in review).

19.4 REFERENCES

Ford, Bacon and Davis, Utah, Inc., 1979, Engineering Evaluation of the Former Middlesex Sampling Plant and Associated Properties, Middlesex, New Jersey: Salt Lake City, Utah, Final Report, FBDU-230-001, prepared for the U.S. Department of Energy.

Environmental Analysis of the Former Middlesex Sampling Plant and Associated Properties, Middlesex, New Jersey: Salt Lake City, Utah, Final Report, FBDU-230-001, prepared for the U.S. Department of Energy (1979).

W. W. Holman and A. R. Jumikis, "Engineering Soil Survey of New Jersey-Middlesex County," Rutgers University Bureau Engineering Research Report No. 10 (1953).

National Lead of Ohio, Inc., Phase I Remedial Action of Properties Associated with the Former Middlesex Sampling Plant Site: Cincinnati, Ohio, Project Report NCRO-006EV, prepared for the U.S. Department of Energy (1982).

W. D. Nichols, "Ground Water Resources of Essex County, New Jersey," New Jersey Division Water Resources Special Report No. 28 (1968).

F. B. Van Houten, "Late Triassic Newark Group, North-Central New Jersey and Adjacent Pennsylvania and New York," in Seymour Subitzky (ed.), Geology of Selected Areas in New Jersey and Eastern Pennsylvania, and Guidebook of Excursions: Geol. Soc. America Guidebook, Atlantic City Meetings, pp. 314-47 (1969).

J. Vecchioli, L. D. Carswell, and H. F. Kasabach, "Occurrence and Movement of Ground Water in the Brunswick Shale at a Site near Trenton, New Jersey," U.S. Geol. Survey Prof. Paper 650-B, pp. B154-B157 (1969).

Roy F. Weston, Inc., "Hydrogeology of the Former Middlesex
Sampling Plant Site, Middlesex, New Jersey - West Chester,
Pennsylvania," Final Report prepared for Oak Ridge National
Laboratory.

CONVENTIONAL REACTOR PROGRAM

20. THREE MILE ISLAND PROGRAM

A. P. Malinauskas

20.1 INTRODUCTION

Activities associated with the Three Mile Island (TMI) Program were of two types. One involved providing technical review and guidance for specific recovery efforts at TMI, whereas the second was concerned more directly with providing technical assistance to recovery operations through detailed analyses and experimental activities. The work was divided into four elements: Task I — coordination of and participation in the operation of the Technical Advisory Group (TAG) for the cleanup of aqueous streams at TMI; Task II — participation in the Technical Assistance and Advisory Group (TAAG) on TMI operations; Task III — chemical development and other technical support to TMI recovery operations; and Task IV — development of inorganic sorbents for the decontamination of aqueous streams. At the program review that was conducted approximately mid-fiscal year, it was decided to curtail the Task IV activities in favor of studies of more urgent problems.

20.2 PROGRESS

Technical progress for each of the tasks of this program for the fiscal year just ended follows.

20.2.1 Technical Advisory Group Activities

The TAG activities were concerned primarily with the prediction and subsequent analysis of the operation of the Submerged Demineralizer System (SDS) for cleanup of the high-activity-level water in the containment

building and for the adaptation of the SDS to decontaminate the reactor coolant system (RCS) water.

A number of lesser activities involved considerations of the following problems: (1) processing of the containment building water head; (2) elution of radiocesium from an in-containment decontamination experiment utilizing a pushcart ion-exchange column; (3) the disposition of slightly contaminated ammonium hydroxide solution; (4) methods for the removal of chloride from RCS water; (5) the use of sodium nitrate or nitrate to suppress radiolytic hydrogen production in SDS ion exchange columns; (6) the employment of decontamination reagents in the reactor building and their potential effect on SDS operation; (7) the need to design a canal/reactor vessel cleanup system for use in defueling; and (8) analysis of concentration changes during feed/bleed of RCS water to ascertain the extent of mixing in the primary system.

20.2.2 Technical Assistance and Advisory Group Activities

Unlike TAG, whose activities are generally restricted to matters involving aqueous stream cleanup, TAAG considerations extend to all TMI recovery efforts. The Group, in fact, played a major role in the axial power shaping rod motion tests and the "quick look" video camera insertions during the past year. TAAG reports formally to the TMI-2 Plant Manager after each series of deliberations. These reports are available through the EG&G Technical Interpretation Office at TMI.

20.2.3 Technical Support Activities

The technical support activities generally consist of a number of ad hoc operations of relatively short duration, which are conducted to resolve

specific problems that arise during the recovery efforts. The work is, however, generic in the sense that similar problems would be encountered in any light water reactor (LWR) accident recovery effort. During this fiscal year, the technical support work consisted of the following activities.

20.2.3.1 Expected Source Terms During Defueling Operations

This activity consisted in convening an in-house peer review group to evaluate a document entitled "Estimated Source Term for Radionuclides and Suspended Particulates During TMI-2 Defueling Operations." The group concluded that the source terms were grossly overestimated. A second draft of the document is currently in preparation.

20.2.3.2 TMI Criticality Safety Support Study

Calculational criticality studies were conducted on various postulated mixtures of fuel pellets, fuel "fines," water, and boron to develop information that is required in assessments of proposed recovery procedures. The results of the studies were transmitted in a report entitled Nuclear Criticality Safety Studies of Interest to TMI-2 Recovery Operations, ORNL/CSD/TM-193 (October 1982) by J. T. Thomas.

20.2.3.3 Steam Generator Tube Corrosion

Inadvertent injection of thiosulfate into the TMI-1 RCS resulted in extensive stress corrosion cracking of the steam generator tubes. Assistance was provided to General Public Utilities at U.S. Department of Energy request to ascertain the nature and cause of the corrosion. Guidance in the development of a research program aimed both at identifying the specific cause of the cracking and at developing a means of preventing non-through-wall cracks from propagating was also provided.

20.2.3.4 Elution of Pushcart Zeolite

In order to remove the pushcart column that was installed in the TMI-2 containment building for use in preliminary testing of the SDS flowsheet, it was advantageous to decontaminate the zeolite that was employed in the test and was thereby loaded with radiocesium.

An experimental study was conducted to develop an elutriant which would introduce no foreign chemicals into the containment building and moreover could be used in a straightforward manner. The use of cesium borate solution was shown experimentally to provide a viable solution to the problem.

20.2.3.5 Methods for Chloride Removal from RCS Water

A series of experiments was conducted to determine the capability to reduce the concentration of chloride ion in the RCS water from the reported 0.7-2 ppm level to a level of 0.1 ppm. Materials selected for testing included silver-loaded zeolites, borate-form anion exchange resins, silver-loaded cation exchange resin, and silver metal powder. The silver-loaded zeolites and the silver-loaded cation exchange resin were shown to be attractive candidates for chloride removal, provided problems associated with silver chloride precipitation could be overcome.

20.2.3.6 Thermal Failure Mechanisms of Self-Powered Neutron Detectors

A series of experiments was conducted with four objectives in mind: (1) determine if self-powered neutron detectors (SPNDs) can fail through simply thermal-activated processes; (2) determine the temperature or temperature range of failure; (3) determine the thermal limits for thermal activation of a SPND; and (4) postulate a failure mechanism for the SPNDs.

at TMI. The tests, which were conducted to temperatures of 1300°C, led to the conclusion that the thermal response of the SPND was dominated by effects to the electrical leads. Although some chemical interaction between the leaks and the electrical insulation was noticed, the tests to 1300°C did not lead to shorting of either cable or sensor.

20.2.3.7 Sorption Rate Studies

Studies were made of the contact time required for each of two specially selected zeolites to absorb either cesium or strontium from solutions simulating those to be expected to result from TMI-2 reactor building surface washdown operations. Results of the tests demonstrated a clear superiority of one of the zeolites over the other for early sorption of both cesium and strontium. The results also indicated no pronounced dependence upon temperature over the range 24-54°C for the preferred zeolite.

20.2.3.8 Analysis of RCS Water and Containment Building Sludge

A sample of RCS water that was taken at 0700 hours on June 14, 1982, was analyzed chemically. Boron isotopic analysis indicated the naturally-occurring distribution; this resolved a question that had arisen prior to the axial power shaping rod motion tests. Uranium isotopic analysis indicated 2.34% ^{235}U and 97.59% ^{238}U at a uranium concentration of 1.9×10^{-4} $\mu\text{g/mL}$. Other species analyzed were at more or less expected concentration levels. The carbonate concentration was found to be 0.04 molar, which tends to confirm the likelihood that fission product strontium that escaped from the fuel during the accident subsequently precipitated as the carbonate.

A sample of containment building sludge is currently undergoing chemical analysis. Preliminary results using spark source mass spectrometry indicate very high concentrations of copper, boron, sodium, aluminum, and nickel, and significant amounts of iron, calcium, chloride, magnesium, silver, and cadmium to be present in the sludge. (Silicon and phosphorous could not be determined by the method employed.)

20.2.4 Development of Inorganic Sorbents

Screening tests involving eighteen inorganic sorbents were made to evaluate this ability to remove cesium and strontium from aqueous systems. These tests identified a number of attractive candidate materials, but the work did not proceed beyond the screening experiments in order to focus on some urgent issues which were identified in the course of the TMI-2 recovery operation.

20.3 PUBLICATIONS

None.

20.4 REFERENCES

None.

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