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**Technical Progress Report for the Quarter
1 October — 31 December, 1980**

Office of Nuclear Waste Isolation

Battelle Memorial Institute
505 King Avenue
Columbus, OH 43201

ONWI
Office of Nuclear Waste Isolation
Battelle

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Office of Nuclear Waste Isolation

**Battelle Memorial Institute
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Columbus, OH 43201**

**Prepared for the
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OVERVIEW

BY

Neal E. Carter
General Manager

The ONWI Technical Program

This report describes the technical accomplishments during the period from September through December, 1980, on the commercial nuclear waste management programs under the direction of the Office of Nuclear Waste Isolation (ONWI). These programs are described in the Report ONWI-19 (Rev. 2), ONWI Technical Program Plan, December, 1980 [see Public Information Section for availability of this document and the previous Technical Progress Report, ONWI-9(4)].

The National Waste Terminal Storage program (NWTs) was established in 1976 and given the responsibility of developing a system for the permanent isolation of nuclear wastes. NWTs is part of the Office of Nuclear Waste Management in the U.S. Department of Energy. The establishment of NWTs signaled the acceleration of the national program.

The purpose of the NWTs program is to develop the technology and facilities required for the disposal of commercial spent nuclear fuel, defense high-level waste, and transuranic waste, ensuring public health and safety and protection of the environment. The program includes facilities for receiving, encapsulation, and disposal of wastes.

The scope of the NWTs repository development effort includes comprehensive geologic exploration, technology development, equipment testing, and model verification activities, as well as the design, licensing, construction, operation, and decommissioning activities required to provide geologic repositories and spent fuel handling and packaging facilities. Development of technology, methodology, and facilities for interim spent fuel storage, waste treatment, and waste transportation, which are interfacing activities with the terminal isolation activity in the national waste management program, are being carried out in other DOE efforts.

ONWI has lead responsibilities in the NWTs structure for oversight and coordination of all programs, for development of the general technology, and for geologic exploration on non-DOE land. A project management group, ONWI directs and evaluates studies, research, and development performed by subcontractors on repository sites and designs.

The scope of ONWI activities includes providing the technology and facilities for the terminal isolation of these wastes by disposal in mined repositories deep underground in stable geologic formations. Steps leading to the accomplishment of that purpose include:

- Site exploration, characterization, and recommendation
- Design, licensing, construction, and operation of a commercial geologic repository (or repositories)
- Technology development to support these steps.

Throughout the process, ONWI has the responsibility to provide public information on all aspects of the program and to encourage public interaction.

It is recognized that safety of the public is the overriding program objective. The technical programs have been formulated to meet the challenge of the NRC licensing process, which is the institutional mechanism for assuring that public protection has been accomplished.

The ONWI program is organized into eight tasks, each one being a level-two element in the NWTS program standard Work Breakdown Structure (WBS).

WBS Element	Task Title
1.1	Systems
1.2	Waste Package
1.3	Site
1.4	Repository
1.5	Regulatory and Institutional
1.6	Test Facilities and Excavations
1.7	Land Acquisition
1.8	Program Management

The purpose of the *Systems Task* is to transform the statement of the ONWI mission into a set of objectives and requirements (the base line) for the ONWI program and to allocate these to each of the remaining tasks. The Systems Task defines the overall strategies and alternatives that will be considered in proceeding toward the goal. It defines subsystems and their interfaces and develops the criteria to be used to determine whether the requirements of the subsystems have been met. It also provides an integrated assessment of the predicted system performance to assure that the overall system requirements have been met.

Conceptually, the geologic repository is viewed as a waste isolation system made up of three functionally distinct but interacting subsystems which provide multiple natural and man-made barriers to the release of the contained waste into the biosphere. These subsystems correspond to the first three tasks: the waste package, the site, and the repository (Figure 1).

The *Waste Package Subsystem* (or Task) is an important part of the overall waste disposal system. During the repository operational phase, the waste package provides safe containment for the waste material during handling and emplacement operations and helps ensure that the waste can be safely retrieved from the repository, if necessary.

In the long term, the waste package provides containment during the thermal period (the time the fission products are decaying), and provides redundant, long-term waste isolation capabilities beyond the thermal period. The waste package will include the waste form itself and a system of engineered barriers that may consist of filler material, a canister, and one or more layers of protective materials selected to minimize interactions among the waste, the host rock, and any ground water that might enter the repository.

The *Site Subsystem* (or Task) includes natural barriers which provide waste containment and isolation. These barriers will keep radionuclides from reaching people in unacceptable

FIGURE 1. GEOLOGIC DISPOSAL SYSTEM

quantities by (1) maintaining the waste in its emplaced position for a very long period of time (i.e., providing waste containment); (2) limiting radionuclide mobility through the geohydrologic environment to the biosphere (i.e., providing isolation); and (3) making inadvertent human intrusion difficult by locating the repository deep in a host rock. The site will contain a host rock suitable for construction of the repository and containment of the waste, as well as surrounding rock formations which can provide adequate isolation. The associated hydrologic system will exhibit low ground-water flow rates, long flow-path lengths to the biosphere, and evidence of long-term stability. The important natural attributes of the host rock will include its hydraulic and chemical characteristics, which determine ground-water flow rates and the dispersion rates of any dissolved waste products; and its thermal-mechanical characteristics, which determine the stability of the host rock to withstand the thermal effects produced by the decay of the fission products in HLW.

The Repository Subsystem (or Task) (Figure 2) will consist of the man-made structures that permit access to the underground facilities and enhance waste containment, and natural

FIGURE 2. WASTE ISOLATION FACILITY CONCEPT

barriers, such as the local host rock, that provide containment and isolation after closure. The design, construction, and operation of the repository will be carried out in a manner that preserves the containment and isolation capabilities of the particular site. Surface facilities will provide for waste receipt, preparation of the waste for emplacement, and transfer of the waste to the underground workings. Repository facilities at depth will include a receiving area for transfer of waste packages to the emplacement area and into the emplacement holes; and equipment to emplace auxiliary barriers, backfill, and other shielding.

The Regulatory and Institutional Task serves to obtain the regulatory, social, and political acceptance of the proposed repository system, without which the mission could not succeed. This task develops criteria that will assure protection of the public health and safety and the environment, and provides the documentation required by the NRC for its review and action before a license to construct is issued to DOE. Not only is public review of the proposed repository project a part of the licensing process, but the DOE has an extensive program to provide all data developed by the program to the public. In addition, under the provisions of the National Environmental Policy Act, all major actions contemplated by DOE will be documented and subject to public review.

The Test Facilities and Excavations Task provides facilities, both in the laboratory and in the field, required to provide physical data and verification of models to be used in design and licensing applications. The acquisition of test facilities requires long lead times, and their need must be justified far in advance of actual operation. Their use will extend beyond the technology phase, being required during design and the subsequent licensing process, to evaluate or demonstrate operability, safety, and reliability. The in situ field facilities allow an experimental integration of the repository subsystems in actual demonstrations of the interactions between the waste package, the repository, and the site.

The Land Acquisition Task will enable preferred sites to remain inviolate after they have been chosen and will see that proper legal procedures are followed in the acquisition process.

The programs required are enumerated in the Work Breakdown Structure (WBS) shown in Figure 3. Progress on various elements of this Work Breakdown Structure is reported in this Quarterly Report in accordance with the WBS elements.

Quarterly Highlights

Major accomplishments during this quarter are listed below.

Systems

- The Disposal Charge Analysis project completed analysis of the cost impacts of storing defense wastes in commercial repositories. It was concluded that the impact of defense waste disposal on spent fuel disposal fees is relatively minor even if large volumes are stored.

- Work continued in several areas of the Radionuclide Migration in Geologic Media project: migration of two-dimensional flow, migration of a daughter nuclide with a solubility limit, perturbation solution of nonequilibrium transport of two species, and migration of

FIGURE 3. ONWI WORK BREAKDOWN STRUCTURE

radionuclides through fractured media. In computer program development, work was completed on a program based on a general analytical solution of nuclide migration downstream from a repository in a one-dimensional ground-water flow.

- One objective this quarter in the Assessment of Effectiveness of Geologic Isolation Systems (AEGIS) project was to provide improved methods for the mathematical modeling of both ground-water hydrologic systems and the transport of contaminants. One of the more significant results was an exact analytical solution for the flow field associated with two fractures intersecting at right angles.

Waste Package

- The major activities during this reporting period for the development of backfill material have been: (1) the acquisition of experimental quantities of candidate backfill materials, (2) continuation of radionuclide distribution coefficients with the materials, (3) initiation of compaction tests, and (4) design and construction of the permeability cell systems.

- Stress corrosion testing of TiCode-12 sheet and rod samples has been extended to more extreme environments in an attempt to identify conditions in which the material shows microscopic evidence of stress corrosion cracking.

- Candidate barrier element materials are being evaluated by means of corrosion screening tests, using Hanford ground water at 250 C.

- A dynamic autoclave facility is being constructed to expose specimens simultaneously to gamma radiation and high-temperature corrosive environments. A slow-strain-rate test facility also is being constructed to determine the susceptibility of metals to embrittlement in repository environments.

- The review of the geologic literature in the effort to anticipate near-field repository behavior has focused upon two major aspects this quarter. The first is the construction and application of various petrogenetic models to the problem of near-field reactions. A petrogenetic model attempts to predict stable mineral phases under a wide range of P-T-X conditions, with emphasis on defining trends in mineral stability under changing environmental conditions. The second is the consideration of the major environmental factors that influence the stability of various potential radionuclide-bearing zeolite minerals.

- A study to assess the effects of several spent fuel waste form alternatives on repository waste package designs was initiated. The study is one of three complementary projects intended to enable the selection of the most appropriate spent fuel form for repository disposal. The study is proceeding by considering several waste package concepts developed for intact spent fuel assemblies and the impact on their design caused by changing the waste form to each of four alternatives.

Site

- Monitoring of acoustic events in the Climax Spent Fuel Test shows that acoustic activity has decreased to less than one event per week. These events are believed to be associated with the thermal expansion of the rock mass caused by heat from the waste canisters.

- The WRIT program continued study of solution mechanisms of doped glass by comparing the results of the M-IAEA and static leach tests. Also, studies of the corrosion of ancient Roman glass show a depletion of sodium to a depth of 3 to 5 microns and calcium to a depth of 1 to 2 microns.

- The source term for spent fuel in a basalt repository developed by WRIT is being used in support of a reference site initial analysis by Pacific Northwest Laboratories. Also, the initial phase of a geochemical modeling study of the basalt ground-water system has been completed. Results indicate that waters are near equilibrium in silica and oversaturated in clay minerals.

- The Gibson Dome No. 1 (GD-1) boring in the Paradox Salt Basin reached a depth of 6,090 feet after a whipstock program successfully bypassed a stuck core barrel. Geotechnical drill-stem tests in GD-1 included five successful loading and hydraulic fracture tests. Work on a conceptual flow model involved defining the flow system for three hydrostratigraphic units located below the salt, within the Paradox Formation, and above the salt.

- Logging of the microstratigraphy of the halite beds in the Gibson Dome core has been completed by the USGS. Geologic cross sections compiled by the USGS through Elk Ridge show that faulting in the Paradox Member is present in the Kidd well (reverse) and the Kubat well (normal).

- The total area in the Permian Basin for area characterization studies is being reduced by application of appropriate screening specifications. A final report distributing the effects of climate on erosion in the Rolling Plains and along the Cap Escarpment was published.

Repository

- At Avery Island, the accelerated borehole closure experiments became operative, and the in-mine portion of the brine movement post-test evaluation began. Also, the extended permeability measurement test holes were drilled, and preliminary measurements were made.

- Laboratory measurements of rock behavior under elevated pressure and temperature at simulated, deep in situ conditions continue. Sample tests run this quarter were marked by exceptionally good behavior of the experimental system and, along with the thermal properties data gathered, allowed a careful analysis of the sensitivity of the data to slight changes in temperature conditions.

- Field experiments in the Stripa Mine, Sweden, have yielded data on the thermomechanical response of a granite rock mass to the heat that might be produced by buried nuclear waste. The materials properties test program continues to study and acquire thermomechanical properties using drill core (small and ultra-large) from Stripa.

- The international Stripa testing program was initiated during this quarter. This program will include hydrogeological investigations in boreholes, migration in single fissure (preparatory study), and a buffer mass test.

- The heated block tests in granitic gneiss are used to determine the in situ mechanical, thermal, and hydrologic properties of a jointed crystalline rock under conditions of varying temperature and stress. The main activity this quarter was to perform three different load-path cycles under ambient (12 C) conditions. The resultant effects of joint closure, opening, and shear are clearly reflected in the results.

- The program to obtain experimental data on the wellbore damage zone caused by full-scale drilling operations has been extended to include five rock types: anhydrite, salt, granite, basalt, and tuff. These additions were made to obtain fundamental data for the design of sealing systems in each host rock type. Tests conducted on anhydrites confirm the data that were collected in the field at the Bell Canyon Test site. The tests have conclusively shown that all flow has occurred around the plug, i.e., between the cement plug's outer wall and the side wall of the hole. Consequently, we now have an established method for testing sealing designs in the laboratory.

- During this quarter, the codes SPECTROM and STEALTH were compared using a room-scale benchmark problem. The SPECTROM programs continue to be developed and verified as required to provide state-of-the-art analysis capability for rock mechanics problems unique to the ONWI program.

- A project to analyze the thermal/mechanical/hydrological response of rock was initiated. This project will further develop and verify the STEALTH computer code as it applies to geologic isolation of nuclear wastes. The project will also perform generic design studies that will form the design and licensing data base for site-specific analyses.

Regulatory and Institutional

- Representatives of the NRC, DOE, and ONWI met to discuss the NWTS approach to model development and the relative scope of the NWTS and NRC model development programs. The meeting resulted in a better understanding by both groups of the model development efforts presently under way.

Test Facilities and Excavation

- The Salt Test Facility (STF) Preconceptual Design/Cost Estimate project was organized and initiated. In the area of granite test facilities, a detailed and comprehensive program plan, Alternate Geology Test Facility Mine Screening, was submitted for review and comment by NPO.

Major Public Information Events of the Quarter

Significant activities were undertaken during the October to December period in support of DOE efforts to provide information on the NWTS program to state and local groups. In addition, there continued to be considerable work in the development and showing of exhibits, in the use and development of audiovisual materials, and in the provision of materials and information for use by representatives of the press, radio, and television.

On November 12, a public meeting on the NWTS program was held in Salt Lake City, Utah. Among the materials prepared for this meeting was a compilation of questions and answers on nuclear waste management. Extensive collections of NWTS reports were made available to all Utah libraries as a meeting follow-up. Briefing packages, including discussions of issues and answers, were prepared for briefings on the NWTS program in Maine and Vermont. A proposed briefing package was prepared to explain the expansion of geologic studies through the DOE-USGS Basin Range and Province Screening Program.

Materials submitted to the Department of Energy for review also included a new leaflet, *Illustrated Mechanics of Nuclear Waste Isolation*; eight fact sheets; news releases on the evaluation of Gulf interior salt domes and on the ONWI annual meeting, including a television news clip; and an NWTS background information pamphlet.

Final plans for museum and condensed versions of a major exhibit on nuclear waste isolation were submitted for DOE review. The plans were developed for ONWI under contract with the Columbus Center of Science and Industry (COSI). ONWI's exhibits calendar had several significant entries during the quarter:

October 22-24, Society of Mining Engineers, Minneapolis

November 16-19, American Nuclear Society, Washington

November 16-20, American Institute of Chemical Engineers, Chicago

November 17-20, Geologic Society of America, Atlanta

December 9-11, NWTS Annual Meeting, Columbus.

A new film on the NWTS program, *Nuclear Waste Isolation Progress Report*, was pre-viewed at the 1980 NWTS annual meeting. This film is designed to complement the existing two-projector ONWI multi-image presentation by showing the various field and laboratory investigations which support the mined geologic disposal concept. An edited version of the film is expected to be available in the spring of 1981. Meanwhile, the two-projector presentation continued to have wide public showings at events such as the following:

A news seminar on nuclear waste disposal in Portland, Oregon, October 23-24

A series of events during October and November sponsored by the Westinghouse Nuclear Center, Pittsburgh.

The November 12, Salt Lake City, Utah, public meeting

A November 6 briefing for railroad executives in Chicago.

Major Reports Published During the Quarter

ONWI-35	<i>Borehole Plugging Telemetry System (IRT/Develco)</i>
ONWI-110	<i>Standardized Repository and Encapsulation Facility Cost Estimates for Comparative Evaluation and Pricing Study (Battelle Project Management Division)</i>
ONWI-115	<i>Parametric Study Involving Thermo/Viscoelastic Analyses of a Room and Pillar Configuration (RE/SPEC, Inc.)</i>
ONWI-128	<i>Waste Isolation Safeguards and Security System (Allied-General Nuclear Services)</i>
ONWI-191	<i>Projected Costs for Deep Geologic Repositories for Spent Fuel Disposal (Battelle Project Management Division)</i>

- ONWI-212 *Annual Proceedings of the National Waste Terminal Storage Information Meeting (Battelle Project Management Division)*
- DP-1533 *Earthquake Related Displacement Fields near Underground Facilities (Savannah River Laboratory)*
- LBL-8313 *Instrument Evalution, Calibration, and Installation for the Heater Experiments at Stripa (Lawrence Berkeley Laboratory)*
- LBL-9392/SAC-26 *Some Results from a Field Investigation of Thermo-Mechanical Loading of a Rock Mass When Heaters Are Emplaced in the Rock (Lawrence Berkeley Laboratory)*
- LBL-10559/SAC-27 *Progress with Field Investigations at Stripa (Lawrence Berkeley Laboratory)*
- LBL-11204 *Proceedings of a Workshop on Thermomechanical-Hydrochemical Modeling for a Hardrock Waste Repository (Lawrence Berkeley Laboratory)*

TECHNICAL PROGRESS REPORTS

In this section the progress reports of the programs in the various waste management technologies are presented according to the organization of the Work Breakdown Structure (WBS). Each report carries a WBS number for the convenience of the reader. The content of the second level of the WBS follows:

WBS 1.1 Systems. These reports deal with systems analysis, socioeconomic assessment, alternative disposal concepts, and performance assessment.

WBS 1.2 Waste Package. These reports consist of studies on performance measures, waste forms, barrier materials, the design and testing of waste packages, and performance evaluation through modeling, demonstration testing, and natural experiments.

WBS 1.3 Site. These reports deal with the earth sciences of geophysics, geochemistry, and transport properties, and with geologic, hydrologic, and environmental characterization of sites that provide natural barriers in salt, granite, and shale.

WBS 1.4 Repository. The repository reports involve studies to build a data base; equipment, instrument, and seal development; repository engineering and conceptual design; and performance evaluation.

WBS 1.5 Regulatory and Institutional. The work reported in this area concerns the regulatory, social, and political acceptance of the proposed repository system.

WBS 1.6 Test Facilities and Excavations. This area is concerned with facilities in both salt and alternative media which will eventually be used for laboratory and in situ testing of the relationships among the package, repository, and site.

SUMMARY

1.1 SYSTEMS

The scope of the Systems Task covers the systems analyses, socioeconomic assessments, and performance assessments of the entire waste isolation system. Also covered in this task are investigations of alternative disposal concepts for nuclear waste disposal. The prime objective of this task is to identify the options available to resolve the problems of waste isolation and to make suitable choices among the alternatives by broad ranging, multidisciplinary analyses.

During this quarter, systems studies continued to develop an analytic overview of the ONWI program. In the Systems Analysis subtask, major emphasis was on analyzing the issues which determine the organization, economics, and safety of the entire repository system. Socioeconomic studies stressed the fiscal interactions between local governments and repositories, while alternative concept evaluations focused upon the analysis of the very deep hole concept. The Performance Assessment subtask was active in studies of radionuclide migration and development of improved modeling techniques. Specific tasks are described below.

Work was completed on the Systems Characteristics Analysis project and a final draft report, *Uncertainties of Sensitivities in the Performance of Geologic Nuclear Waste Isolation Systems*, was issued.

Work on the Waste Isolation Program Evaluation Assistance project included three tasks. Management-by-Decision (MBD) charts were prepared, emphasizing effects of interactions with the NRC licensing process. An analysis of proposed Congressional legislation on nuclear waste was performed to identify potential changes in the NWTs programs. The third task, initiated this quarter, was a study of Intermediate Scale Facilities and their potential impact on the NWTs programs.

The Disposal Charge Analysis project completed analysis of the cost impacts of storing defense wastes in commercial repositories. It was concluded that the impact of defense waste disposal on spent fuel disposal fees is relatively minor even if large volumes are stored.

The Systems Safety and Risk Assessment Evaluation project concluded that the differences in estimates of risk for handling, storage, and transportation are largely attributable to accident probability values and cesium release fraction estimates.

Detailed charts were prepared this quarter on which the final report for the Performance Measures Development project will be based.

A draft report, currently in the ORNL review process, on Fiscal Interactions Between Local Governments and Federally Owned Waste Isolation Facilities, was completed as part of the Institutional Arrangements for Radioactive Waste Siting project. The report contains an analysis of transfer payments from federal facilities to affected communities and concludes that several payment strategies are available.

In developing systems descriptions for the very deep hole concept, it was necessary to redefine the concept of primary containment to include the VDH repository host rock zone. Although this concept does not preclude containment within the engineered waste package

during the time of primary fission product decay, it highlights the uncertainties of assuring such containment at VDH depths. Conversely, the expanse of host rock surrounding the waste package at depth is considered a barrier not available to shallower repositories.

An interim program plan was prepared for research to assess the properties for host rock media relevant to very deep hole waste isolation as part of work in the Very Deep Hole Geosciences project.

A demonstration example was prepared to illustrate the Very Deep Hole Systems Decision Studies project methodology. This example illustrates the approach developed by SRI International for evaluating major research and development funding decisions pertinent to VDH waste disposal research and development planning.

Work continued in several areas of the Radionuclide Migration in Geologic Media project: migration of two-dimensional flow, migration of a daughter nuclide with a solubility limit, perturbation solution of a nonequilibrium transport of two species, and migration of radionuclides through fractured media. In computer program development, work was completed on a program on a general analytical solution of nuclide migration downstream from a repository in a one-dimensional ground-water flow.

Acquisition, conversion, integration, and operational testing of existing modeling technology for the SCEPTER project continues. One accomplishment this quarter was the operation and verification testing of a ground-water flow model for fractured media.

One objective this quarter in the AEGIS project was to provide improved methods for mathematical modeling of both ground-water hydrologic systems and the transport of contaminants. Of the results reported, one of the more significant is that an exact analytical solution was found for the flow field associated with two fractures intersecting at right angles. Details of the work are provided in the quarterly.

WBS 1.1.2

Project: Waste Isolation Program Evaluation Assistance

Principal Investigator: Lawrence Livermore Laboratory (R. C. Maninger)

ONWI Project Manager: J. D. Waddell

Objective

The objective of this project is to enhance the ONWI capability for NWTS program planning. Specifically, the project shall:

- Design a methodology for representing the planned structure of the research and development activities and results of the NWTS program.
- Develop a methodology for coalescing the data being generated by the NWTS program and presenting the results in a form most useful to decision makers.

Progress Reported Previously

This project was started in March, 1980. During the period July to September, 1980, two sets of Management by Decision (MBD) charts were prepared to represent major parts of the NWTS program from two different perspectives. This illustrated the importance of having well-defined purposes for constructing MBD charts and of using no more graphic detail than necessary to show essential elements and their contextual relationships for each purpose. Adjacency matrices were also introduced as another way to show contextual relationships among elements of an MBD representation of a plan. The matrices complement the charts and are useful as aids in preparing charts.

Activities During the Reporting Period

MBD representations of the nuclear waste repository licensing process, as proposed by NRC, were prepared. The purposes of these charts are to give a concise portrayal of the proposed process, to focus on definitions of critical terms, and to aid in coupling the NWTS technical program to the NRC licensing program.

Also, several Congressional bills on nuclear waste were examined from an MBD perspective and charts were prepared. The analysis of the Congressional bills was used to identify potential changes in the NWTS program. Finally, a study of the potential impact of intermediate scale facilities (ISF) on planning and institutional aspects of the NWTS program was started.

The NWTS program is a major part of a much broader National Nuclear Waste Management program. NWTS is concerned with the DOE responsibilities for safe, deep, geologic disposal of commercial and defense high-level wastes. These responsibilities include site selection, technology development, repository design, repository construction, and waste management system operation. Other agencies also have responsibilities in the overall

national program. For example, NRC is responsible for licensing repositories to receive and dispose of high-level radioactive waste. The NRC intends to carry out its responsibilities with a plan described in the proposed rule for licensing procedures (10CFR60, 1979).

In the NRC view, the licensing of a repository will proceed in four consecutive stages, each addressing the same technical issues but with progressively increasing knowledge leading to increased confidence that HLW can be disposed of safely. In MBD language, the four stages are considered as four broadly aggregated and contextually related activities, including site characterization and three reviews: (1) license application for construction authorization, (2) amendments to license application to receive wastes, and (3) repository activities and application to decommission (NRC). Table 1 shows the definitions of importance and the statements of expectations of the NRC.

The activities of DOE and NRC intertwine in the licensing process. Strategic contingency paths exist for cases in which decisions are rendered against progressing to following stages. These paths are not indicated by 10CFR60 (1979), but are suggested by logical analysis of the overall waste management program. They emphasize the interactions of DOE and NRC decisions and their impacts on time schedules and costs. (Each pass through an activity adds delay time and costs.)

Two different legislative approaches have been formulated to define and emphasize national nuclear waste management needs. The first is a "hardware" approach, with emphasis on unlicensed R&D demonstration repositories for reprocessed waste. The repositories would be designed for future expansion into full-scale repositories licensed by NRC. Even though the demonstration repositories would not be licensed, there would be substantial consultation and coordination with state and local officials in states where repositories would be located.

The second is a very different legislative view of critical waste management needs: a regulatory and institutional approach with very detailed definitions of the roles of Congress, states, Indian tribes, the Executive Branch, and the NRC in the selection of sites and regulatory licensing activities up to the time that a construction authorization is granted or denied. There are essentially no technical directives. Only full-scale repositories are considered, but the possibility for exploratory shafts under NRC guidance is provided.

Current proposed legislative bills are not expected to become law; they do reveal some of the thinking in Congress on waste management issues. Their overall purpose was to expedite action in the National Nuclear Waste Management program. The role of the NRC in licensing is subject to wide ranges of scope among the bills. In all cases, full-scale repositories would be licensed. However, demonstration repositories, test facilities, early shafts, etc., could be constructed without licensing by NRC, depending upon the particular legislation finally approved by the Congress and the President. It is also possible that reprocessing of spent fuel will be reinstated so that spent fuel itself will not be the waste form for disposal. New legislation may also call for and define changes from current policies in the relative importance assigned to isolation by the waste package (engineered barriers) versus the site and regional geology (natural barriers). Any of these issues can lead to substantial changes in the structure, cost, and scheduling of the NWTS program. General analysis of the impacts of these broad issues is beyond the scope of this particular project.

TABLE 1. DEFINITIONS AND STATEMENTS OF EXPECTATIONS OF THE NRC

Site Characterization	Commencement of Construction	Decommissioning
<ol style="list-style-type: none"> 1. "Site characterization" means the program of exploration and research, both in the laboratory and in the field, undertaken to establish the geologic conditions and ranges of those parameters of a particular site relevant to the procedures under this part. Site characterization includes borings, surface excavations, excavation of exploratory shafts, limited subsurface lateral excavations and borings, and in situ testing needed to determine the suitability of the site for a geologic repository, but does not include preliminary borings and geophysical testing needed to decide whether site characterization should be undertaken. 2. To satisfy requirements of NEPA, the NRC anticipates such characterization at a minimum of three sites representing a minimum of two geologic media. However, in light of the significance of the decision of selecting a site for a repository, the commission fully expects the department to submit a wider range of alternatives than the minimum suggested here. 	<ol style="list-style-type: none"> 1. Commencement of construction means clearing of land, surface, or subsurface excavation, or other substantial action that would adversely affect the environment of a site, but does not include changes desirable for the temporary use of the land for public recreational uses, site characterization activities, other pre-construction monitoring, and investigation necessary to establish background information related to the suitability of a site or to the protection of environmental values, or procurement or manufacture of components of the geologic repository operations area. 2. The submittal of a site characterization report by the department not only begins the commission's involvement in the planning and development of a geologic repository, but also marks the beginning of state participation in the licensing process. 	<ol style="list-style-type: none"> 1. "Decommissioning" means final backfilling of subsurface facilities, scaling of shafts, and decontamination and dismantlement of surface facilities. 2. "Disposal" means permanent emplacement within a storage space with no intent to retrieve for resource values. 3. Unless expressly authorized in the license to receive and possess HLW, an amendment to that license will be required to allow the department to conduct partial backfilling in parts of the repository once all the wastes have been emplaced in those parts. (This does not apply to backfilling tests that are described in the license.)

A study was begun on the narrow and specific issue of intermediate scale facilities as part of the NWTs program based on various Interagency Review Group, House, Senate, and DOE documents.

Since the only documents that refer explicitly to intermediate scale facilities are the IRG reports, the IRG definition of an ISF has been adopted. Table 2 summarizes some major points on ISF as contained in IRG (1979).

In the definition of ISF, the ambiguity of meaning of spent fuel assemblies versus waste canisters could have major impact on cost and scheduling. The IRG report does not contain any references to the time and cost impacts of ISF on the regulatory activities of the NRC.

If the ISF is to be finished in time to contribute some technical information to full-scale facility activities, the licensing procedures for the ISF must require less time than for a full-scale facility and thus be somewhat less strict. How much less? Another set of licensing criteria, regulations, procedures, etc. will probably be needed for ISF. Thus, any prospective savings in ISF licensing time could be greatly exceeded by the ISF regulation promulgation process. None of the Congressional bills that were studied passed in the 96th Congress, but similar

TABLE 2. INTERPRETATION OF IRG AND RECOMMENDATIONS ON ISF

1.0 Definition of ISF:	A facility in which some hundreds, perhaps as many as 1,000 spent fuel assemblies or waste canisters would be emplaced. Waste would be emplaced with the possibility of removing it, if necessary, but without expectation to do so.
2.0 Purposes of ISF:	2.1 Acquiring technical, engineering, and operational data. 2.2 Exercising and learning about licensing and organizational processes. 2.3 Using results from 2.1 and 2.2 to aid in the transition from R&D to full-scale permanent operational repository facilities.
3.0 Necessity for ISF:	3.1 ISF can play a distinct and desirable role in the transition from R&D to full-scale operational repositories.
4.0 Recommended Policy and Constraints on ISF:	4.1 The ISF shall be licensed. 4.2 Regulations and criteria set by NRC shall prevail for licensing. 4.3 The ISF shall not delay the pace of the program leading to the first full-scale repository. 4.4 The ISF shall be built only on a schedule that is sufficiently ahead of the opening of the first full-scale repository to contribute to some significant aspect of the waste management program. 4.5 The ISF shall be undertaken only if there is adequate technical readiness and an adequate site characterization has been completed. 4.6 The ISF shall be a stand-alone facility or colocated with a TRU facility.

efforts will probably emerge from the 97th Congress. All of them were designed to supply specific policy guidance, to remove institutional uncertainties, and to speed up the process of getting safe operational repositories.

Many issues are raised by the ideas on intermediate scale facilities and related concepts that must be resolved primarily through institutional and executive policy formulation rather than technical developments:

- Will NRC use different sets of criteria for ISF and full-scale facilities?
- What is adequate for technical readiness and site characterization for an ISF versus a full-scale facility?
- Where should ISF be located with respect to full-scale repositories?

- How shall ISF impacts on schedules be determined?
- Will waste package or geologic media be the primary barrier?
- Will ISF contain spent fuel, solidified HLW, or both?

The time required to resolve these issues depends on the degrees of risk the top-level policy makers will take, given the current state of knowledge of nuclear waste problems and the public pressure for action.

Future Activities

In the next quarter, the institutional and nontechnical analysis of ISF and alternatives will be coordinated with technical analyses and completed. Actions of the 97th Congress on nuclear waste management issues will be monitored and analyzed by MBD methods to identify potential impacts on the NWTs program.

REFERENCES

10CFR60, 1979. *Federal Register*, Vol. 44, No. 2136, p. 70408, 6 December.

Interagency Review Group on Nuclear Waste Management, 1979. Report to the President, March, pp. 54-65.

WBS 1.1.2

Project: Disposal Charge Analysis

Principal Investigator: Battelle Pacific Northwest Laboratories (L. L. Clark)

ONWI Project Manager: J. D. Waddell

Objective

The objectives of this study are (1) to analyze the impact on spent fuel disposal charges of isolating defense wastes in a commercial spent fuel repository and (2) to estimate the costs of disposing of defense wastes in this manner.

Progress Reported Previously

Preliminary estimates were made of the impacts on repository receiving rates, area requirements, and costs if defense wastes are put in commercial spent fuel repositories. Current estimates were made of high-level and transuranic waste inventories accumulated to the year 2000. The conceptual repository scenario arbitrarily assumed that three regionally located repositories in bedded salt will come on-line near the year 2000 and substantial unused handling capacity will exist for receipt and disposal of defense wastes.

Repository costs were estimated and allocated to wastes according to the relative numbers of containers emplaced or the ratio of mined volume by waste type, depending on the cost category. For the high-volume case, a larger portion of spent fuel costs was allocated to defense wastes because of the larger defense waste volumes. The Hanford cesium and strontium wastes have large unit disposal costs because of their high-heat loads and hence wide spacing in the repository. The other high-level wastes have significantly lower unit cost since their emplacement density is not heat-limited. Disposal of remote-handled TRU wastes costs significantly more per drum than disposal of contract-handled TRU wastes because of the different handling and emplacement requirements.

It was concluded that the impact of defense waste disposal on spent fuel disposal fees is minor even if larger volumes are terminally stored.

TABLE 1. ESTIMATED DEFENSE WASTE INVENTORIES AND DISPOSAL VOLUMES FOR WASTES ACCUMULATED TO THE YEAR 2000

Waste Type	Year 2000 Inventory	Processed Volume Sent to Repository	
		Reference Case	High-Volume Case
High-Level Waste	315,000	35,500	235,000
TRU Waste	924,000	74,200	425,000

Activities During the Reporting Period

Work was completed on the analysis of the cost impacts of storing defense wastes in a commercial spent fuel repository. The results of this analysis showed that such an option would cost on the order of \$100 million to \$2 billion in 1980 dollars, depending on the degree of volume reduction in waste treatment and the extent to which defense wastes are recovered. Factors that significantly affect the change were determined to be:

- Discount rate provisions
- Uncertainty in expenditure projections
- Timing of expenditures and collection of fees
- Whether or not an additional contingency factor is included
- Institutional factors.

Other factors that affect the disposal charge less significantly are the leveling period over which the charge is calculated, cost allocation mechanisms to distribute costs related to away-from-reactor storage and disposal, and allocation of R&D and other overhead costs to other transuranic wastes besides spent fuel.

Future Activities

As a result of the analysis, it was determined that the discount rate provision and institutional effects should be studied in more detail, since these may be subject to controversy. At the same time, data will be gathered on the cost variances as a function of geologic media.

WBS 1.1.2

Project: Systems Characteristics Analysis, a Summary

Principal Investigator: Lawrence Livermore Laboratory (W. G. Sutcliffe)

ONWI Project Manager: J. D. Waddell

Objective

The objective of this project is to develop information on how the uncertainties about engineered and natural systems affect the uncertainty of long-term performance for nuclear waste repositories. The purpose for developing this information is to provide background information which ONWI can use in formulating R&D plans. This work provides a systems analysis perspective.

The primary objective is to identify the uncertainties dominating the uncertainty about the performance of a system. In particular, the conditions under which various subsystems and parameters dominate will be identified. Also, the potential differences between waste forms, geologic media, and site characteristics and how these differences affect the assessed performance of the system will be considered.

The scope of the work has been limited to postclosure analysis of nuclear waste migration via ground-water flow.

Progress Reported Previously

Since this project was initiated in May 1980, the work has concentrated on developing and exercising systems models in probabilistic analysis of geologic waste isolation systems. Previous quarterly reports presented the simple models used to characterize the waste form and package, repository site, and regional aquifer subsystems. The methods of analysis and computation were presented with examples of quantitative results. The computations were nearly completed, and the report for this study was being drafted.

Activities During the Reporting Period**Uncertainty Analysis**

During this period, the analysis was completed, the results were interpreted, and conclusions were drawn for the report. The final draft of this report, entitled *Uncertainties of Sensitivities in the Performance of Geologic Nuclear Waste Isolation Systems*, was completed.

The study consisted of a parametric analysis of the overall systems model and an analysis of a hypothetical system using detailed models of the subsystems. The purpose was to investigate the relative effects of the uncertainty about each model parameter and determine the general characteristics of systems which can be represented by a point source and series pathway model.

Both deterministic and probabilistic parametric analyses were conducted over a range of parameter values large enough to encompass significantly different sites in different media,

and different waste forms and packages. These results are applicable in general to isolation systems which can be represented by the model. The parameters of the high-level model were the transit time, the waste pulse spread for the system, and the dissolution time. The systems analysis considered more fundamental parameters of the waste migration pathways, treating these parameters probabilistically. The analysis demonstrated the contribution of each subsystem to overall system performance.

Conclusions were drawn about the system model parameters, relative to the particular systems and scenarios that were analyzed. In addition, some of the general conclusions are summarized below:

- Global parametric analysis is more informative and applicable than local sensitivity analysis.
- The individual contributions of each of the "multiple" barriers (subsystems) are often additive rather than multiplicative.
- The uncertainties associated with the weaker subsystems (for a given scenario) have little effect on the uncertainty in system performance.

Short-Term Study

In addition to the ongoing systems uncertainty analysis, a short-term study on Intermediate Scale Facilities was initiated and completed this quarter. In particular, the need for information about repositories before they are designed, constructed, and operated led to the suggestion that a Repository Test and Training Facility be constructed.

Future Activities

Extensions or applications of the work include: (1) the study of additional scenarios, (2) a site-specific study (such as the Spent Fuel Site at NTS), (3) an analysis of correlations between parameters, and (4) a detailed analysis of subsystems (e.g., a repository scale study to determine the effect of various designs and emplacement plans).

Possible future activities include: (1) a general study of waste isolation systems to define and classify types, functions, constraints, alternatives, criteria, subsystems, and parameters; (2) study of repository operations and systems; and (3) a feasibility study for a Repository Test and Training Facility.

WBS 1.1.2**Project:** Very Deep Hole Systems Decision Studies**Principal Investigator:** SRI International (F. L. Offensend)**ONWI Project Manager:** R. E. Best**Objective**

The objective of this project is to develop a pilot decision model for use in managing the Very Deep Hole research program. The model is to be demonstrated with a preliminary evaluation of selected research strategies, including a preliminary comparison of the very deep hole (VDH) concept with the mined geologic repository (MGR). The expectation is that many of the planning principles developed for the VDH program will be transferable to other R&D programs within ONWI.

Progress Reported Previously

The project was initiated 7 April 1980. Most of the development work on the project was completed prior to the current reporting period. Methodology was developed during the first portion of the project to show how research decisions and outcomes affect VDH system performance. A simple model of the mined geologic repository was also developed. Comparing the projected performance of MGR and VDH as a function of different possible VDH research results provides a basis for identifying the VDH research strategies that might enable VDH to enhance or complement MGR. Comparing the cost of the VDH research with the value that society assigns to the enhanced MGR system then provides a basis for judging whether the VDH research strategy is cost-justifiable.

Activities During the Reporting Period

Most of the work during the current period was spent preparing a demonstration example of the decision methodology and preparing a draft final report. The demonstration example considers a hypothetical decision on whether to fund a \$1 billion research program on VDH. The example assumes that four different research projects on permeability have been proposed to facilitate the major funding decision. The immediate decision addressed by the examples is to decide which, if any, of the proposed projects to fund.

The four projects are evaluated with respect to how they facilitate the major funding decision. Based on the illustrative data and value assignments used, it is found that, without additional information, it is cost-justifiable to fund the VDH research effort, even though there is an 82 percent chance that the results of the effort will not contribute to the final design of the nuclear waste program. This decision changes given certain of the possible results of the four proposed projects. Taking into account the likelihood and reliability of these results, as well as the fact that they would lead to saving the \$1 billion expenditure, the values of the four different projects are calculated.

The illustrative calculations indicate an \$11 million value for conductivity studies and zero value for certain theoretical studies on fracture formation and for direct observation studies in deep boreholes being drilled for other purposes. However, funding the conductivity studies and theoretical studies together gives a net value of \$30 million. The reason for the increase in value is that if the two studies are performed together, and if they verify each other, then there is sufficient basis to change the assessment on ground-water velocity, making it easier to judge the viability of VDH.

Future Activities

ONWI comments on the draft final report are expected. A final report will be prepared after responding to the ONWI comments.

WBS 1.1.3

Project: Institutional Arrangements for Radioactive Waste Siting

Principal Investigator: Oak Ridge National Laboratory (E. Peele)

ONWI Project Manager: J. R. Finley

Objective

The objective of this project is to assess the impacts of nuclear waste isolation facilities on local and state host areas and to design siting arrangements which are acceptable to local interests.

In particular, the program is examining ways in which communities may be compensated for the usual and unusual costs imposed by siting. Usual costs refer to the public service needs of the facility and its staff juxtaposed with the circumstance that because the facility is federally owned, it is not liable for state and local taxes. Unusual costs refer to the fact that waste facilities of this type may be perceived by local residents as imposing risks in excess of typical manufacturing enterprise.

Several potential devices to compensate communities for these two types of costs are currently under examination, with primary efforts focused on payments-in-lieu-of-taxes (PILOT). Ultimately, the project will evaluate these measures and report the advantages and disadvantages of each.

Progress Reported Previously

This project was initiated during the previous quarter. Three background papers on existing institutional arrangements were begun: Fiscal Interactions Between Local Governments and Federally Owned Waste Isolation Facilities, Review of Existing Institutional Arrangements and Incentives for Siting Publicly Sensitive Facilities, and Evaluation of the Payments-in-Lieu-of-Taxes Mechanism for Rad-Waste Communities.

Activities During the Reporting Period

Economic analysis during the reporting period led to the completion of the draft Fiscal Interactions Between Local Governments and Federally Owned Waste Isolation Facilities, which presents an analysis of transfer payments from a federally owned waste isolation facility to local governments affected by the facility. Two types of payments (PILOT or those community and block grants to other affected communities) are examined, and it is concluded that several payment strategies are available. Some could provide very substantial incentives for communities to encourage siting.

A distinction is made between the "compensation of costs" and the "provision of incentives": Unless a community can be shown to reach a higher level of satisfaction, under a rather strict interpretation of satisfaction, it will have no incentive to encourage the siting of a facility, whether or not the direct costs associated with the facility are compensated. Despite this fact, it should not be difficult to design an intergovernmental revenue system that does provide incentives.

These findings rest on a number of assumptions and use a number of behavioral suppositions that must be validated and measured before such systems can be implemented. First, it is assumed that payments-in-lieu-of-taxes based on a property tax equivalency scheme is indeed a feasible alternative. If past DOE policies, such as those used to encourage "self-sufficiency" in "atomic communities", underlie the development of the PILOT schedule, it is unlikely that any incentives to site will be generated. Second, it is assumed that waste facilities inherently contain elements of risk that must be compensated. If levels of risk can be minimized through siting schemes or design, the role played by transfer payments is reduced, and the design of the payment system is simplified.

Third, it is assumed that communities are willing to trade off the risk associated with waste facilities for transfer payments. Clearly, if this assumption is erroneous, or subject to special preconditions or qualifications, the probability of creating incentives is greatly reduced or perhaps no longer relevant.

Two sets of behavioral parameters are also highly important in successfully designing an incentive system based on transfer payments. The first regards the community response to the price change which occurs when a large increase in taxable capacity per worker is added to the tax roll. Estimating this parameter will require careful examination of the range of local governments in which the facility might be sited and will yield a range of payments estimated.

The second important parameter governs the rate at which communities value the trade-off between risk and compensatory payments. Again, the estimation of this parameter will require careful examination of communities likely to be affected and will yield estimates of the magnitude of a transfer payment program to compensate or extend costs.

Future Activities

Work scheduled for next quarter includes the following: (1) an analysis of the revenue requirements of a property tax-based PILOT system will be started; (2) preliminary research on an economic fiscal analysis model will begin; (3) existing PILOT systems will be evaluated; and (4) the evaluation of existing siting arrangements for sensitive facilities will be continued, via literature review and interview techniques.

WBS 1.1.4

Project: Nuclear Waste Management Space Option

Principal Investigator: NASA Marshall Space Flight Center (MSFC) (W. E. Galloway)

ONWI Project Manager: R. E. Best

Objective

The overall program objective is to assess the desirability of using a space option to dispose of certain high-level nuclear waste as a complement to the currently planned national nuclear waste management program. The specific technical objectives of this year's activity are to: (1) identify reasonable alternative concepts for space systems; (2) provide a preliminary definition of each concept; (3) develop a methodology for the concept selection process; (4) evaluate the concepts as to their technical viability; (5) select attractive space system concepts for further consideration; (6) document the evaluation and selection process; and (7) provide appropriate engineering, reliability, environmental impact, and cost data for input to DOE activities on the space option.

Progress Reported Previously

This project was initiated on 1 June 1979 under a NASA/DOE Interagency agreement. Current status of knowledge on the space disposal of nuclear waste as identified in the past and by current studies is as follows: (1) space disposal of nuclear waste is technically feasible with current and projected space transportation systems and waste forms; (2) space disposal of *entire spent fuel elements* from commercial plants and *all* of the highly diluted defense waste will not be practical (excessive number of launches) with space transportation technology that is likely to be available by the year 2000, and therefore a radionuclide removable process will be required; (3) current technology nuclear waste forms (glass and calcine) are not practical waste forms for space disposal, but the ORNL iron/nickel/copper-based cermet waste form appears suitable; (4) various launch systems and destinations have been investigated.

Activities During the Reporting Period

Activities included (1) mission analysis, (2) waste payload systems, (3) flight support systems, (4) launch site systems, (5) orbit transfer systems, and (6) systems integration and evaluation.

Mission Analysis. Efforts were devoted to defining rescue mission requirements. Abort modes were described which would result in a safe orbit for the waste payload in the event of an injection stage failure. The performance of the delivery vehicles was determined to be sufficient to meet rescue mission requirements.

Waste Payload Systems. The results of the trade studies have been used to define waste payload configurations for the total system options. Configurations defined include integrally shielded waste payloads (nonremovable shields) and waste payloads with radiation shields that can be removed in low earth orbit or at earth escape.

Flight Support Systems. Flight support system requirements established previously plus the waste payload configurations already described were used to define the flight support systems. Efforts concentrated on accommodating the "hardened" waste payload options by not using an ejectable waste payload protection system. Systems were defined capable of supporting both single and dual waste payloads in the Space Transportation System Orbiter cargo bay.

Launch Site Systems. The Kennedy Space Center's (KSC) impact of the selected alternative systems was assessed as approximately equivalent to a 60-launch-per-year rate. Additional facilities required were calculated assuming that they are dedicated to the space disposal mission. These amount to three Orbiter Processing Facility bays, three Vertical Assembly Building cells, four Mobile Launch Pads, and two launch pads.

Alternative sites were assessed using inputs from past studies. The major reason for selecting a remote site is isolation in event of a catastrophic launch pad accident. Past studies have investigated the viability of remote sites, but when combined with the results of the waste payload system task, indications are that this objective can be better met by proper design of the waste payload protective system.

Orbit Transfer Systems. Efforts concentrated on defining (1) the orbit transfer system options identified previously and (2) the rescue systems. Options were defined by generating point designs for candidate injection and placement stages, including cryogenic injection and placement stages, single-stage Lunar Landing Orbit Transfer Vehicles (LLOTV), storable placement stages, and solar electric single-stage-to-destination-and-placement stages.

A radiation effects assessment of the injection and placement stage avionics was conducted to determine the effects of the intense radiation flux from an unshielded waste payload. Results indicate that, for nominal mission profiles, placing the avionics so that they are shielded by the fuel tanks of the Solar Orbit Insertion Stage (SOIS) is sufficient protection. To allow for the 3-year length of rescue missions and for contingencies involving loss of propellant, approximately 770 kg of shadow shielding has been added to all SOIS stages used with unshielded payloads to protect rescue-critical avionics.

The results of this study indicate that by using a single-injection stage and one or two SOIS vehicles, all rescue mission performance requirements can be met.

Systems Integration and Evaluation. Evaluations were conducted to integrate the results of the waste payload system, flight support system, launch vehicle, and orbit transfer system tasks into total system concepts and to trade the concepts defined to select those offering the best combinations of cost and risk.

Twelve candidate total system concepts were defined incorporating all feasible combinations of launch, orbit transfer, and waste payload protection options. Concepts were evaluated for performance expressed in dollars per kilogram of waste payload delivered to 0.85 astronomical units (AU). Values for all options were then normalized to the updated reference system. The 12 options showed normalized costs between 64 and 240 percent of the cost associated with the reference concept. Options were then ranked for risk by comparing the risk-reduction features (such as waste payload ejection or hardening) associated with each concept.

Four concepts were selected for having costs equivalent to or lower than the updated reference and also showing lower risk.

The four concepts identified as attractive alternatives include: two dual launch concepts which carry the fully shielded waste payload all the way to 0.85 AU, offering reduced risk at costs of 65 to 76 percent of the reference system cost. Two single-launch solar electric stage concepts which also carry the fully shielded waste payload to 0.85 AU were identified.

Future Activities

Activities planned will include the following:

- Completion of the final oral and written reports for the 1980 study.
- Initiation of two FY 81 studies.
 1. *Space System Follow-On Study*—The objective is to define major impacts on the reference space systems concepts (as defined by the FY 80 study) that would result from changes in the reference waste mix.
 2. *Risk Model Assessment Study*—The objective is to develop an approach and methodology leading to the development of a space systems risk model for the nuclear waste disposal mission.

WBS 1.1.4

Project: Very Deep Hole Geosciences

Principal Investigator: Lawrence Berkeley Laboratory (LBL) (L. Cohen and T. Doe)

ONWI Project Manager: R. E. Best

Objective

This project is aimed at further review and assessment of geological, hydrological, and geochemical uncertainties to improve the state of knowledge and provide bases for a program to define and develop the very deep hole (VDH) waste disposal alternative.

Progress Reported Previously

During the previous quarter, a long-range, comprehensive deep hole research plan was prepared. It covered the basic areas of numerical modeling, laboratory experimentation, and field tests in holes drilled for other purposes but available for research.

A report on the availability of *Experimental Opportunity Holes* was aimed at providing the preparation and lead time needed for resolving access and mobilization problems associated with conducting research in holes drilled by other government and private agencies. Early results indicated that considerable possibilities existed, even at depths in excess of 3,000 meters, and that valuable testing could be done at much lower expense than would be the case with a dedicated test hole.

Activities During the Reporting Period

A reduced level, interim program plan was prepared for deep hole research pursuant to executive policies de-emphasizing alternatives to mined repositories. The interim plan consists of:

- Numerical modeling of a deep hole repository
- Laboratory experiments on borehole failure and fracture closure
- Monitoring technology and research relevant to very deep hole disposal
- Preparation of a new long-range program plan.

Arrangements were made for studies on borehole failure modes and fracture closure in granite, shale, and salts at high temperatures and pressure at the U.C. Berkeley campus.

In November, personnel from the Danish Electric Utilities, who are responsible for the Danish waste storage effort, visited LBL and exchanged information on their plans for use of deep hole repositories in salt, and on their current research in a 3,400-meter hole in domal salt in Jutland.

Future Activities

The data base document will be sent to ONWI for review on 31 January 1981. Laboratory experiments will be initiated. Work on the revised program plan and technology monitoring will continue.

WBS 1.1.4

Project: Very Deep Hole Engineering Studies

Principal Investigator: Woodward-Clyde Consultants (C. L. Taylor)

ONWI Project Manager: R. E. Best

Objective

The objective is to perform a planning study for the very deep hole disposal concept which will identify the significant problem areas, identify and discuss attributes of the VDH disposal concept, narrow discussions to a reference VDH concept, produce preliminary cost estimates, provide a program plan for VDH facilities and equipment development, and provide a rationale for additional work by ONWI if such work is warranted.

Progress Reported Previously

Work was initiated 1 July 1980. The work to date has included a review of previous VDH studies, current drilling technology, and conditions likely to be found in deep geologic environments. Based on these data preliminary generic performance, safety, and design criteria were developed for a VDH system. Contained within the generic criteria are criteria necessary to establish minimum and maximum depths. The next step was to select a VDH reference system to be used in this planning study.

Various VDH schemes were identified for possible use in the reference system and then evaluated as to their particular advantages and disadvantages. Based on this evaluation, the reference VDH system that is envisioned for this study consists of an array of rotary drilled holes advanced to depths of 6,000 meters within a granite host rock. The waste canisters would be placed in a modified casing package and lowered as stacked units into an uncased emplacement zone which is 3,000 meters in height. Following emplacement, each stack would be grouted in place. The borehole would then be sealed above the emplacement zone.

Activities During the Reporting Period

It was decided to reformat and expand the generic criteria under sections addressed to the major subsystems of the VDH disposal system: natural (site) system, waste package, and repository design and operation. The criteria were "tested" against the seven generic performance objectives for NWTS disposal systems outlined in DOE/NE-0007 *Waste Confidence Rulemaking*.

The restructuring of the criteria focused attention on the potential difficulties in a deep hole environment of achieving Objective 1 (Containment) within the generally inferred geometry (within the waste package). For purposes of the VDH system, it was decided to redefine the geometry of containment to include the repository zone, while not precluding containment within the waste package.

The design and costs of the VDH system are impacted by the ability to drill vertical holes with a high degree of accuracy; the amount of casing and mud that need to be left in holes between drilling and emplacement activity (possibly for several years); and the time sequence and distance between drilled holes and active drilling, emplacement holes and active drilling, and sealed holes and active drilling. Based on these various considerations, the present reference system is proposing to use large enough hole spacing to allow emplacement immediately after drilling rather than use closer hole spacing which would require a high degree of drilling accuracy and which possibly would result in a lapse of several years between the active drilling of a hole and emplacement of the waste.

Future Activities

Preliminary cost and a schedule for the VDH system will be developed, along with an engineering program plan. The results of the VDH engineering studies will be presented in a draft report to ONWI within the next quarterly reporting period.

WBS 1.1.5

Project: Assessment of Effectiveness of Geologic Isolation Systems (AEGIS)

Principal Investigator: Pacific Northwest Laboratory (F. H. Dove)

ONWI Project Manager: J. Kircher

Introduction

The overall objectives of the Assessment of Effectiveness of Geologic Isolation Systems (AEGIS) program are to: (1) develop the capabilities needed to assess the postclosure safety of geologic repositories; (2) demonstrate the assessment capabilities by performing analyses of reference sites; (3) apply the assessment methodology to assist the National Waste Terminal Storage Program in site selection; and (4) perform repository site analyses responsive to the time schedule and to the level of sophistication required to meet the licensing needs of the National Waste Terminal Storage Program.

The scope of AEGIS in the past has been limited to postclosure, far-field, long-term analysis. It excluded the consideration of (1) processes that are induced by the presence of the wastes that may affect the repository integrity, and (2) nuclear waste disposal options other than geologic isolation. The scope will be gradually expanded to incorporate near-field, short-term aspects of geologic isolation into a comprehensive performance assessment methodology. In FY 81, AEGIS will continue research and development efforts for assessment capability including the assimilation of near-field models assembled by Intera and will use existing methodology to perform reference site initial assessments, site selections, and other ONWI and DOE support activities.

To date, the reference site initial assessments (RSIA) have been methodology demonstrations, and they will include completion of the salt dome RSIA and continuation of the basalt RSIA using the Geologic Simulation Model (GSM) to analyze geologic processes and events. The geologic simulation code will be applied to model a hypothetical repository in the Paradox Basin as a demonstration of code flexibility in the simulation of a bedded salt environment. The potential application of the geologic simulation code to a hypothetical repository in Canadian granite will further extend the code usability to another geologic terrain.

AEGIS hydrologic, contaminant transport, and dose models will be used to support in situ experiments at the Nevada Test Site and to evaluate the basis for EPA draft standards. These applications represent an awareness that AEGIS methodology, which has been primarily developed for repository licensing activities, has immediate usefulness in other technical areas of the NWTs program. The usefulness of the methodology as a whole or in part is expected to increase in future years.

In FY 81, the AEGIS program is again divided into two functional branches—one for research and development of the performance assessment methodology, and the other for application of the existing methodology prepared for site qualification and licensing. This structure is reflective of the evolution of the AEGIS program as it continues to become more applications oriented. Maintenance of the methodology development work, as a separate branch from the applications work, allows continual research and development without inter-

ference by the immediate demands of site assessments. Additionally, maintenance of an applications group helps to focus the preparation of site assessments and other methodology applications.

This report covers the technical activities of AEGIS for the first quarter of FY 81.

Methodology Development

Objective

The objectives are to provide improved methods for the mathematical modeling of ground-water hydrologic systems and the transport of contaminants, including radionuclides in subsurface systems. The improved methods are needed both as analytical tools during design of a waste repository and for long-term safety assessment, because they will assure that the most effective, advanced, and best evaluation capability is available to apply in analyzing waste repositories in deep geologic media.

Progress Reported Previously

Previous accomplishments in flow methodology development provide (1) an equivalent integral form (Arnold, 1980) for the hydraulic conductivity inverse problem, (2) boundary integral type descriptions for flow in fractured permeable media, and (3) a potential theoretical basis for approaching the solution of the generally ill-posed permeability increase problems.

The concise, closed-form mathematical description of the flow field in a fractured porous material involving a single fracture was completed. The purpose in deriving both the single-fracture and the intersecting-fracture flow fields is to provide precise representation of these two basic building blocks of a fractured system in order to synthesize very complicated fracture systems statistically; thus, the scale-related aspects of an equivalent porous medium and the related dispersive phenomenological effects can be exhaustively studied.

Progress in the contaminant transport methodology development area centered around providing contaminant transport analysis under uncertainty and using geohydrologic response functions to analyze overall repository system performance. The extended operator equations and an uncertainty analysis method (Gibbs, 1980) form the basis for the current uncertainty research and developmental work. Response functions were found to efficiently summarize the many interacting effects contributing to the overall performance of a repository system. Their simplicity and completeness also make them useful in assuring communication between the technical disciplines and the decision maker or general public.

Activities During the Reporting Period

Flow Methodology Development

Work on the hydraulic conductivity inverse subtask was directed at an integral form of the inverse problem which hopefully gives smoother results with respect to input velocity fields than previous methods. A candidate numerical procedure for this problem was formulated following a variational principle (Arnold, 1980). Based upon these results, the numerical procedure should be an improvement over present techniques, but it is not expected to alleviate all ill-posedness problems.

Developmental work on fluid flow under uncertainty brought about an overall recognition of the problem being somewhat more complicated than that of uncertainty in transport. Specifically, the storage or time derivative term of the flow equation contains stochastic information in the form of an imprecisely known storativity. Use of the finite element technique gives rise to a matrix multiplying the vector of known potentials and leading to more complex matrices with complicated products and triple products.

The continuing work toward describing flow in a fractured permeable medium provided significant results. An exact analytical solution was found for the flow field associated with two fractures intersecting at right angles, and the two fractures in the porous material had different fracture widths and lengths, resulting in an unequal flux redistribution at the intersection.

An approximate analytic solution for two intersecting fractures meeting at any angle was also derived that requires representation of the intersecting fractures by more segments than if a completely closed-form expression were applied. However, comparison with the 90 degree exact solution gives excellent agreement when using more segments in the approximation. Exact solutions were also obtained for intersection angles that can be represented as the quotient to two integers, allowing the required integrals to be evaluated exactly.

Detailed development is under way to provide rapid and direct solution capability for very complicated combinations of intersecting and nonintersecting fractures. The input and output routines are completed, allowing interactive construction of complex fracture patterns and output of the resulting flow system potentials and streamlines. An important part of this capability is providing ways to handle fractures that intersect flow boundaries. Encouraging progress has been made in handling such intersections with rectangular flow boundaries.

Transport Methodology Development

Significant progress has been made in applying geostatistics to determine the hydraulic conductivity and ground-water potential data covariance structure by geostatistically modeling the field data from the Hanford Reservation. The drift and covariance structure of hydraulic conductivity and ground-water potential data are necessary to provide the required pore velocity input uncertainties for the new stochastic convective transport equations and are currently being characterized for use as input to the initial stochastic convective contaminant transport code.

A way was found to overcome the computational problems in the uncertainty transport development work. The problems arise from the more involved numerical solution requirements for stochastic as compared with deterministic transport equations. In classical deterministic numerical solution methods for time-dependent problems, only results at two consecutive times are involved in the stepwise solution process. By contrast, the numerical solution of the uncertainty transport equations involves results from all preceding times in the solution sequence, requiring the formation of powers of large matrices, each arising from a triple vector of matrix products.

Work on this problem previously centered on finding an appropriate approximation for the large matrices to reduce computations to a reasonable level. One approximation tried was the use of an identity matrix where only the main diagonal elements from the large matrices were raised to powers. Difficulty had also been found in numerically calculating the variance with sufficient accuracy. There was really no way to determine if the resultant lack of accuracy was related to the large matrices approximation or to the poorly estimated variance by numerical means.

One method not only simplifies the computations but also permits the use of all terms under some conditions in both the mean concentration and its variance calculation in the time sequence. This is particularly important since the problems of approximating the reciprocal powers of matrices, solution updating, etc., are no longer present.

The simplification, brought about by the simple form the matrix takes for the finite difference approximations, has only been carried out for the one-dimensional case, but the extension to the two-dimensional case is not a problem. It is not clear whether the matrix for the finite element approximation will permit this type of simplification. In the finite difference case, the troublesome summation is replaced with a linear combination of matrices computed recursively from a two-term relation. In the one-dimensional case, there are two such matrices; in the two-dimensional case, there are five.

With the one-dimensional code and analytical solutions (Gibbs, 1980), the use of approximations, such as replacing the reciprocal powers by diagonal matrices, can be tested. Such a comparison for the expectation and the variance has and is being made in some detail. Results so far indicate it is entirely feasible to solve the one-dimensional problem for both the mean concentration and its variance without making any approximations. A similar statement can be made about the two-dimensional case, but it will have a higher cost associated with a solution, and, unless the resulting matrix is simple, some approximations will probably be used. In the case of a finite element formulation with a highly irregular boundary, the matrix would contain too many nonzero elements to consider the approach developed to present. If the boundary is simple, the resulting matrix might permit one to avoid these approximations.

Because the geohydrologic response functions are really overall contaminant inflow-outflow relationships for a particular subsurface repository and associated geohydrologic system, they effectively summarize all the pertinent happenings between the source of contaminant and its exit to the biosphere. Some secondary or side benefits have been realized in their use over and above the direct use as system evaluators that significantly aid in the effective communication of technical results. They are useful in demonstrating or comparing the significance or sensitivity of system parameters or effects on overall repository performance. If a large change in the graph of the response function results from a parameter change, it is immediately apparent. Further, the simplicity in directly evaluating the magnitude

of system change due to a parameter change is very helpful. In each case, it is easy to visualize or see expected effects on the overall system performance as caused by the sensitivity change.

Response functions are also useful in setting up distributed process control approaches to provide more usable repository site selection and utilization procedures. The initial problem is being set up and handled as an adjoint formulation to provide minimal effects to the biosphere by optimal repository placement in a deep regional ground-water flow system.

Problems

The repeated occurrence of triple matrix products in the solution of the uncertainty flow and transport problem emphasizes the need for greater computing capacity in a direct interactive mode.

Future Activities

A formal write-up of the proposed numerical method for the inverse problem will be prepared and revised until numerical testing appears feasible. A difficult problem remaining will then be a mathematical demonstration that the method is convergent, stable, and better behaved with respect to the roughness of input data than previous methods.

Recurrence relationships will be sought for the various terms occurring in the stochastic fluid flow formulation, using two approaches: (1) a more general recurrence formula for matrix products, and (2) the simplistic domain and regular discretization concept. Examination of the form of the algebraic equations resulting from the finite element algorithm may yield simplifications and recurrence relations similar to those found in the contaminant transport case.

Completion of solutions for the integer angular intersection in fracture flow will enable an accurate evaluation of the approximate representation for intersecting cracks. Code development will continue, with particular emphasis upon rapid field solution methods.

A priority item for the immediate future is the insertion of the Hanford data into the error analyses for the perturbation scheme to see if the variances are sufficiently small for the perturbation scheme to be accurate.

Site Applications

Progress Reported Previously

Multiple runs of the Geologic Simulation Model were made in the Monte Carlo mode for testing and debugging purposes.

Sensitivity studies with the RECHARGE code were completed, as was the regional hydrologic model.

Activities During the Reporting Period

Scenario Analysis

The input-output and statistical packages of the Geologic Simulation Model, formerly called the Release Scenario Analysis Model, were completed, and several hundred Monte Carlo simulations were run. Data from these runs were used for logic checkout and debugging. Despite the lack of a complete set of operational input data, the GSM in the Monte Carlo mode was useful in characterizing the effects of uncertainties in knowledge of present ground-water characteristics of the Pasco Basin on the initial modeled ground-water flow system.

Adaptation of the GSM to salt anticlines of the Paradox Basin was initiated. This is intended to be a technology demonstration effort rather than a fully-developed RSIA, and it is expected that the GSM for the Paradox Basin will be less detailed than that for the Columbia Plateau. Efforts during this quarter were limited to a field reconnaissance of the Paradox Basin and reviews of the literature.

Work on the fault-zone permeability and crustal stress-strain studies was also begun, but it was limited to a review of the literature.

Hydrogeologic and Transport Analysis

There is a lack of quantitative data on the regional transmissive properties of the deeper basalts, and extensive efforts are associated with the regional recharge calculations. It should be noted that both the BWIP hydrologic needs for the Pasco Basin and the AEGIS regional and local modeling needs for the basalt RSIA were successfully fulfilled.

All hydrologic and transport data were stored in appropriate computer files, but preliminary analyses with the Pasco Basin FE3DGW flow model were delayed by problems with the hydrologic data. Disparities between structure, potentiometric surface, and transmissivity maps apparently resulted from inconsistent choices of map boundaries during digitizing. These problems, and ones resulting from underestimating the complexity of the modeling task (especially in connection with interfaces between regional and local hydrologic models), delayed completion of the hydrologic and transport modeling for the basalt RSIA by almost two months.

Dose Analysis

Potential radiation doses at one million years after disposal of spent reactor fuel in a salt dome were recalculated using a revised list of radionuclide contents. No significant differences were noted between the latest results and the doses calculated previously. The work will generate a preliminary estimate of radionuclide release standards for comparison with those of the EPA. ONWI authorized completion of the dose code DITTY1, required for calculation of population doses integrated over a 10,000-year period, and performance of the necessary dose calculations. Encoding of the computer program DITTY1 was completed. The program is fairly complex and quite comprehensive with regard to possible scenarios and exposure pathways.

The code is operational on the UNIVAC 1100/44 and has been debugged. However, spot comparisons with the PABLM code occasionally yield different answers. The two codes are currently being reviewed to find an explanation for these differences.

Preliminary calculations of the potential health effects from release of the quantities of radionuclides listed in the EPA Draft HLW Criteria were repeated with two different scenarios during December. The EPA release criteria were designed to result in not more than 1,000 health effects/10,000 yr/100,000 MTHM for a generic repository. The first set of calculations assumed one person directly ingested the entire quantity of a nuclide listed in the EPA criteria table. The resulting doses were then assumed to be population doses (in terms of man-rem). A conversion factor of 2×10^{-4} cancers per man-rem was used to convert whole-body doses to health effects. The resulting health effects ranged from 10^2 to 10^5 for the list of nuclides. The second set of calculations was performed, assuming that the released nuclides were diluted in well water used for irrigation and human and animal consumption for 70 years. When the total diluting volume was taken to be sufficient to serve 200 persons (i.e., 50 wells, 5 gallons/minute each, serving 50 farm families of 4 persons each), the calculated health effects were reduced to 1 to 10 with the exception of ^{99}Tc which had a value of 0.01 health effects. It is very possible that EPA used a more conservative (higher) value for the uptake of ^{99}Tc by plants than the 0.25 used in PNL's code. ORNL had recommended a tentative value of 25 a few years ago, but has since found somewhat lower values in vegetation samples collected in the field.

A test data transfer to DITTY1 from AEGIS geotransport codes was initiated. This work is necessary to develop the links between geologic simulation, hydrogeologic and transport modeling, and dose analysis necessary for an integrated evaluation of a repository.

Future Activities

The Geologic Simulation Model will be used in operational Monte Carlo analyses, assuming that sufficient consultant inputs can be achieved. These analyses will be the basis for evaluation of both the GSM and the design disruptive sequences of a hypothetical repository in Grande Ronde basalts underlying the Hanford Reservation. This will be the first step in operational integration of the geologic simulation/hydrogeologic modeling/dose analysis series for evaluation of repository performance.

The hydrologic and transport modeling for the basalt RSIA is scheduled for completion by mid-January. Sensitivity studies are to follow with results available prior to the 1 April milestone for DOE clearance.

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WBS 1.1.5

Project: Systematic Comprehensive Evaluation of Performance and Total Effectiveness of Repositories (SCEPTER)

Principal Investigator: INTERA Environmental Consultants, Inc. (R. B. Lantz)

ONWI Project Manager: H. C. Burkholder

Objective

The objective of this program is to integrate, develop, and apply technology to evaluate the effectiveness of a nuclear waste isolation system in preventing adverse radiological effects to present and future human beings and their environment. The technology will be used to support technology development, site evaluation, repository design, waste package design, and qualification and licensing activities of the NWTS program.

Progress Reported Previously

The SCEPTER contract began in January, 1980. Work during the first three quarters concentrated on three fundamental areas: (1) program planning, (2) defining the scope of the modeling technology required to support the NWTS program, and (3) reviewing and evaluating the available technology. Initial drafts of the Management Plan, Quality Assurance Manual, and SCEPTER QA Program Plan were submitted. A draft Technology Needs Definition letter report was submitted.

In addition, a one-week workshop on Uncertainty Analysis in Post-Closure Repository Assessment was planned and conducted. The first development activity under the SCEPTER program was initiated: the development of a dual porosity-discrete fracture flow model. The first application activity for the SCEPTER modeling technology was identified: an independent assessment of the LLL and LASL hydrology and nuclide migration tests at the Nevada Test Site.

Activities During the Reporting Period

The SCEPTER program has been restructured into two technical tasks (far-field and near-field analyses). This restructuring reflects the change recommended in the approach to performance assessment—from standard scenario and consequence analyses to modified system analyses. The restructuring affects neither the scope nor the objectives of SCEPTER.

Significant Technical Milestones

Most important to the SCEPTER program at this stage are the acquisition, conversion, integration, and operational testing of existing modeling technology. The integrated structure of a waste isolation system-level model and the structure of subsystem models for the geosphere, biosphere, repository, and waste package have been defined. In each of the subsystem areas, process models have been identified for operational testing. In the far field, some selected geosphere and biosphere models for each of the functions have been acquired, converted, and integrated, and these have undergone initial operational testing. For these functions, other selected models are yet to be obtained. In the near field (repository and waste package), the appropriate existing models have been identified and acquisition and conversion have begun. In addition, the AEGIS far-field scenario model has been

selected as the basis for the SCEPTER far-field state model. This far-field state model integrates the various natural and human-induced phenomena that can modify the biosphere and geosphere in ways that might alter the waste isolation system. The AEGIS model has been acquired and converted, and the appropriate input-output support modules have been developed for implementation on the SCEPTER computers. Operational testing of the far-field state model is awaiting appropriate test data from PNL. These data should be received early in January, 1981.

The draft Technology Needs Definition letter report was expanded in response to comments received from ONWI. This report, *Summary of SCEPTER Modeling Technology Needs*, summarizes and defines the total scope of modeling technology required to support waste isolation system performance assessment within the NWTS program.

A report that documents the results of the review and evaluation of existing modeling technology is nearing completion. This report will be the end product of the review and evaluation subtask under the SCEPTER program. It identifies existing models that should be acquired for operational testing, specifies modifications that should be made, and recommends developing new models where the existing technology falls short.

The development of a ground-water flow model for fractured media continued on schedule. The flow model, which includes both flow in discrete fractures and a dual-porosity representation of flow in the continuum between major fractures, is operational and has undergone verification testing.

Application of SCEPTER's modeling technology in the assessment of data to be gathered during the LLL hydrology and nuclide migration tests at the Nevada Test Site was more clearly defined. Discussions with LLL personnel led to small changes in the design of the initial series of hydrology tests. The initial tests were conducted and the data gathered. After additional testing and processing of data, SCEPTER will perform independent analyses of the tests and assist in the design of the nuclide migration tests.

Comments were received from participants at the Uncertainty Analysis Workshop. These comments will be integrated into the Uncertainty Analysis report that will reflect the status of this important subject.

Future Activities

Activities scheduled for the next quarter include:

- Complete operational testing of existing far-field models
- Continue acquisition, conversion, integration, and operational testing of existing near-field models
- Define the performance specifications for models recommended for development
- Contribute to the NWTS Performance Assessment Plan
- Complete the Uncertainty Analysis report
- Begin assessing the LLL hydrology test data

- Finalize the technology transfer procedure
- Continue developing the ground-water flow model for fracture media.

WBS 1.1.5

Project: Analysis of Radionuclide Migration in Geologic Media

Principal Investigator: University of California at Berkeley (T. H. Pigford)

ONWI Project Manager: H. C. Burkholder

Objective

The objectives are to develop analytical solutions and to make diagnostic calculations of the migration of radionuclides through geologic media. The exact results of these calculations can be used as benchmark checks against numerical computer codes. The solutions are used to predict the space-time-dependent concentrations of radionuclide chains as they migrate through geologic media and to make diagnostic calculations of the effects of phenomena not previously considered or not previously well understood in the modeling of radionuclide transport.

Activities During the Reporting Period**Computer Program Development**

A computer program (UCBNE20), based on the general analytical solution of nuclide migration downstream from a repository in a one-dimensional ground-water flow has been developed. The theory for this is presented in Section 4.4 of the 1979 report (Harada et al). The program is applicable to nuclide chains of arbitrary length in the absence of dispersion.

Migration of Two-Dimensional Flow

The study of nuclide migration in two-dimensional ground-water flow through geologic media is continuing. The analytical solution of the migration in a flow field due to a source-sink combination, as presented in the 1980 report (Pigford et al), has been evaluated. A computer program (UCBNE21) has been developed which yields the concentration-space-time dependence for this type of flow field for nuclide chains of arbitrary length in the absence of dispersion. Migration in other two-dimensional flow fields is being investigated.

Diffusion-Limited Dissolution of Solid Waste

An analysis describing the dissolution rate of a three-dimensional vitreous waste as limited by molecular diffusion in the liquid boundary layer surrounding the waste has been completed. Work on the effects of convective flow on the dissolution rate of waste material is in progress.

Migration of a Daughter Nuclide with a Solubility Limit

In Pigford et al (1980) the one-dimensional transport of a single radionuclide with a solubility limit was analyzed, assuming dissolution from an infinite plane source. The solubility limited concentration occurs at the source, and the solubility limit affects the magnitude, shape, and duration of the chromatographic band as it migrates through the sorbing medium.

There are some nuclide chains for which the solubility limit may apply to a daughter radionuclide which may not be initially present in the waste. Depending upon the local geochemistry, this could occur for thorium in the decay chain $^{234}\text{U} \rightarrow ^{230}\text{Th} \rightarrow ^{226}\text{Ra}$, and the resulting concentration profile of migrating ^{226}Ra would be thereby affected.

Analysis of the migration behavior of a solubility-limited decay daughter is complicated because the region of saturation concentration occurs away from the dissolving waste, and the location and extent of this region varies with time. An approximation to the solution to this moving boundary problem, in the absence of dispersion, has been developed and demonstrated. Studies are under way to develop a more rigorous solution to this problem.

Perturbation Solution of the Nonequilibrium Transport of Two Species

The exact solution of the coupled equations for transport of two nonequilibrium species of a radionuclide through a sorbing medium is presented in Pigford et al (1980). Because that solution involves somewhat complicated functions, a simpler approximate solution by expansion in a power series of the ratio of rate constants has been derived using a perturbation method. The approximate solution gives a fairly good approximation to the exact solution under stationary conditions. A comparison of the approximate solution for two migrating species with the exact solution will be given later.

Migration of Radionuclides Through Fractured Media

Studies have been initiated on nuclides migrating through rock along convective paths in macrofractures, as affected by diffusion into and sorption in local microfractures. Solutions for the following transport cases of a single nuclide are being developed:

- Diffusion-limited penetration into microfractures, local sorption equilibrium
- Rapid diffusion into microfractures, slow sorption-desorption in microfractures
- Penetration into microfractures limited by diffusion rate and sorption rate.

Future Activities

The analysis of radionuclide transport in two-dimensional flow fields is to be extended to include dispersion. Demonstrations will be made of the application of the present analysis to migration in a two-dimensional flow field. Studies will be initiated on the dissolution rate of a solubility limited waste as affected by convection in the liquid boundary layer adjacent to the waste. Work will continue on the consolidation of the computer codes for one-dimensional

migration. Studies will continue on migration in fractured media, with and without local equilibria between solution and solid.

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WBS 1.1.5

Project: Systems Safety and Risk Assessment Evaluation

Principal Investigator: The Analytic Sciences Corporation (J. W. Bartlett and B. Ross)

ONWI Project Manager: J. D. Waddell

Objective

The purpose of this program is to evaluate and integrate the results of current and past safety and risk assessments and to evaluate the advantages and disadvantages of different strategies for applying safety and risk assessment methods to the licensing process.

Progress Reported Previously

A review was made of a wide spectrum of studies whose purpose was to analyze the danger from accidental releases during presealing waste management operations involving the handling, storage, and transportation of spent fuel or high-level solidified waste in the commercial light-water reactor or defense fuel cycles.

The following are the principal results and conclusions of the evaluation and comparative analysis:

- Taken as a whole, the existing studies cover all important accident scenarios and containment failure modes for accidents in presealing operations. However, the depth to which the risk components are analyzed varies considerably from study to study.
- The normalized risk values for handling, storage, and transportation accidents exhibit a spread of several orders of magnitude. This can be attributed largely to variation in accident probability values and cesium release fractions employed in the studies.

A review of the post-emplacement studies presents a variegated picture of the work done in nuclear waste management. Results of safety assessment studies were compared by normalizing the results to a common basis of repository contents at closure and environmental scenario conditions. For the HLW scenarios, the peak doses fall into two natural classes: a class of conservative scoping analyses centered at about 1 percent of natural background radiation and a class of more realistic or less conservative studies centered about five orders of magnitude below background.

The spent fuel studies, like the HLW studies, were consistently below background for credible scenarios. Much lower doses were reported for studies that calculated peak doses within a limited time frame.

The information gained from previous studies on safety may be drawn upon heavily in the formulation of requirements. The identification of scenarios and their concomitant probabilities was identified as a weak area in the majority of past studies. A site-specific analysis should include as complete a list of scenarios as possible, if for no other reason than to address "what if" questions from readers. However, the number of possible repository descriptions may approach infinity. An estimate of probability is necessary to place the consequences of a given scenario in perspective.

Another area that needs to be addressed is the choice of repository performance measures. Future work is strongly affected by how it must address the legal question of safety. The information available from the past work should be used in formulating these legal guidelines.

Activities During the Reporting Period

The studies have been reported in the draft Review of Safety Assessments of Nuclear Waste Management, which reviews past studies of the danger to the general public from the disposal of nuclear wastes in mined repositories. Both presealing operations and long-term dangers after the repository is sealed are considered.

The methods used in presealing studies are relatively well established; however results vary widely because of disagreements in assumptions. The techniques used to assess long-term dangers are more variegated, but there is a considerable degree of agreement in the results.

The fourth chapter, rewritten during the reporting period, is summarized in Table 1.

One factor occurs repeatedly: the differences caused by the change in time frame for pre-and postsealing releases. The limited time frame of the presealing releases implies that:

- Identification of release scenarios may be acceptably complete.
- Quantitative probability estimates are feasible for these scenarios.
- Release scenarios can be sufficiently characterized to obtain an estimate of the consequences within narrow bounds of uncertainty.
- The population will remain relatively stable during the period of exposure.

The incomplete nature of the postsealing p-c (probability x consequence) curve precludes the calculation of an expected dose. Without the ability to complete this curve, it is impossible to place presealing and postsealing safety assessments on a comparable basis.

Future Activities

No further work is planned for this project.

**TABLE 1. COMPARISON OF DIFFERENT ASPECTS OF ESTIMATING
PRE- AND POSTSEALING RISKS**

Aspect	Presealing	Postsealing
Time frame involved	Short-term, immediate future	Long-term, indefinite future
Release mechanisms	Precipitating events	Precipitating events and long-term processes
Identification of release scenarios	Acceptably complete	Complete list may not be feasible
Quantitative probability estimates of scenarios	Obtainable (accident reports, etc.)	Qualitative estimates feasible. Quantitative estimates may not be feasible.
Is an acceptably complete p-c ^(a) curve obtainable?	Yes	Doubtful
Characterization of surrounding "media"	Facilities can be adequately characterized	Characterization of local and regional parameters subject to larger uncertainties
Barriers to release once container is breached	HEPA ^(b) filters, or none (depending on operation)	Multiple engineered and geological barriers
Type of exposure	Acute	Chronic
Consideration of risks borne by future generation	No	Yes
Parameter defining "maximum individual"	Location	Diet
Can the release occur unnoticed?	No	Yes
Potential for cleanup once release is recognized	Yes	No

(a) A p-c curve is a curve of risk (probability x consequences).

(b) HEPA = high efficiency particulate air (filter).

SUMMARY

1.2 WASTE PACKAGE

The scope of the Waste Package Task covers waste form investigations and definitions, engineered barrier materials identification, design and development of waste packages, and the performance assessment of the waste package. The prime objective of this task is to design and develop waste packages that enhance the performance of the total nuclear waste isolation system.

Studies on spent fuel deterioration, waste package characterization, waste form assessment, and disassembly alternatives impact continued this quarter. Data and conclusions from the pretest destructive examination of five rods from the Climax Spent Fuel Test assemblies and the final design and performance of the spent fuel calorimetry system were documented. Fuel assembly structural and fracture mechanics analyses of spent fuel under disposal cycle dynamic loading conditions were completed and were also documented. A perspective on fission gas release from spent fuel during geologic disposal and the examination of the fuel pellet has been completed. The reduction of data from the whole rod test program started this quarter.

The Soret coefficient of NaCl in 3 M NaCl solution between 30 and 60 C has been determined. A new volumetric technique for sampling and analyzing solutions has been developed. This technique is efficient and gives reproducible results.

The major activities during this reporting period for the development of backfill material have been: (1) acquisition of experimental quantities of candidate backfill materials, (2) continuation of radionuclide distribution coefficients with the materials, (3) initiation of compaction tests, and (4) design and construction of the permeability cell systems.

Stress corrosion testing of TiCode-12 sheet and rod samples has been extended to more extreme environments in an attempt to identify conditions in which the material shows microscopic evidence of stress corrosion cracking.

Candidate barrier element materials are being evaluated by means of corrosion screening tests, using Hanford ground water at 250 C.

A dynamic autoclave facility is being constructed to expose specimens simultaneously to gamma radiation and high-temperature corrosive environments. A slow-strain-rate test facility also is being constructed to determine the susceptibility of metals to embrittlement in repository environments.

The NBS Independent Measurement Standards Laboratory is designed to review existing efforts by others to carry out NBS work in selected areas to improve the understanding of leaching. In addition, the effects of radiation, mechanical properties, microstructure, microchemistry, transport mechanisms, and other disruptive processes will be investigated, and support will be provided to the NBS effort on leachability and waste forms.

The Stabilizer Materials Program objectives are to identify, test, select, and qualify stabilizer materials for use in designing waste packages. The preliminary screening of candidate

spent fuel stabilizer materials and the spent fuel stabilizer temperature limit test were completed during this report period. This test was conducted to establish the temperatures below which the melting of metals and metal alloys, fusing of glass stabilizers, or other possible emplacement processes would not compromise fuel rod integrity.

For purposes of screening, the emplacement process temperature limit criterion of 590 C provides a basis for accepting or rejecting stabilizer materials. The immediate impact of this screening criterion on the stabilizer program is the rejection of copper and all its alloys (casting temperature ≥ 954 C) from further consideration as stabilizers for intact spent fuel rod waste packages. The primary function of the stabilizer should be to resist collapse of the canister due to geologic lithostatic and hydrostatic pressures.

The generalized relations of the binary $\text{CaNaPO}_4\text{-Ca}_2\text{SiO}_4$ were reported. Four products were used to start hydration studies: CaNaPO_4 , $(\text{CaNaPO}_4)_{50}(\text{Ca}_2\text{SiO}_4)_{50}$, $(\text{CaNaPO}_4)_{20}(\text{Ca}_2\text{SiO}_4)_{80}$, and Ca_2SiO_4 .

It has been established that high-temperature α -type compounds can be easily quenched from intermediate compositions in the binary $\text{CaNaPO}_4\text{-Ca}_2\text{SiO}_4$. The high-temperature compounds hydrate. Some hydration products incorporate foreign ions, have cementing properties, and have crystallinity, suggesting that they have desirable sorptive properties. The exact manner of the incorporation or sorption is being determined.

The review of the geologic literature in the attempt of anticipating near-field repository behavior has focused upon two major aspects this quarter. The first is the construction and application of various petrogenetic models to the problem of near-field reactions. A petrogenetic model attempts to predict stable mineral phases under a wide range of P-T-X conditions, with emphasis on defining trends in mineral stability under changing environmental conditions. The second is the consideration of the major environmental factors that influence the stability of various potential radionuclide-bearing zeolite minerals.

The chemical and mineral composition of the host rock and of the associated interstitial fluids play a major role in the stability relationships of zeolite and clay minerals. There is relatively good information, from thermodynamic and field analyses, on the occurrences of various zeolite minerals in differing water compositions; this can be visualized on a series of activity-activity stability field diagrams. The influence of host rock composition has proven more difficult to define; problems abound in the chemical similarity of many host rocks that produce vastly different zeolite minerals and also the large variation in composition of presumably similar host rocks.

A study to assess the effects of several spent fuel waste form alternatives on repository waste package designs was initiated. The study is one of three complementary projects intended to enable the selection of the most appropriate spent fuel form for repository disposal. The study is proceeding by considering several waste package concepts developed for intact spent fuel assemblies and the impact on their design caused by changing the waste form to each of four alternatives.

A log of vitrified high-level waste is slated for storage in the Near-Surface Test Facility as part of the Basalt Waste Isolation Project. Preparation of the waste log for shipment is now in progress. The National Lead, Inc. truck cask has been selected for transporting of the waste log to the EMAD Facility.

WBS 1.2.1

Project: Performance Measures Development

Principal Investigator: Arthur D. Little, Inc. (C. R. Hadlock)

ONWI Project Manager: J. D. Waddell

Objective

The objectives of this program are (1) to provide a complete set of performance measures in six categories (environmental impact, societal impact, safeguards, safety, economics, and resource consumption) for use in analyzing NWTS program options and (2) to provide assistance to ONWI in implementing the measures in the NWTS System Evaluation Model.

Progress Reported Previously

The project was started on 21 May 1980. The following accomplishments were reviewed previously: (1) an interim draft report on performance measures, (2) an oral presentation on the performance measures report, and (3) comments submitted on draft NWTS-33 criteria documents (such criteria are related to the performance measures).

Activities During the Reporting Period

Further comments, both general and specific, on draft NWTS-33 criteria documents were submitted.

Based on the charts in the interim draft report, a detailed and uniform set of charts—presenting a logical subdivision of each of the six performance categories—was prepared and submitted in draft form. The final report on performance measures will be based on these charts.

A brief report was submitted on decision considerations for an intermediate-scale facility.

Planning for waste package performance measures development was initiated.

Future Activities

For the next quarter the principal activity will be the development of a detailed program plan for completing the work and submitting a report on what constitutes good waste package performance and what means can be used to verify or assure such performance. This plan will be reviewed with ONWI in mid-March. An extensive literature review will be carried out during the preparation of this plan.

WBS 1.2.2

Project: Spent Fuel Studies

Principal Investigators: Hanford Engineering Development Laboratory (HEDL) (R. J. Cash)

ONWI Project Manager: E. F. Benz

Program Continuation**Objective**

The objective is to complete work that was in progress at the beginning of FY 81 when the program scope was changed at HEDL, including:

1. Document the results obtained from the spent fuel calorimeter system.
2. Complete the structural and fracture mechanics analyses on spent fuel under disposal cycle conditions.
3. Document the effects of geologic disposal tests on the physical, mechanical, and chemical condition of spent fuel rods and assemblies (Climax/Spent Fuel Test).
4. Analyze the results of the whole rod tests.
5. Complete a perspective on the effects of fission gas release from spent fuel rods during geologic disposal.

Progress Reported Previously

Calorimetry of Turkey Point Reactor spent fuel assemblies D15 and D22, destined for emplacement into the Climax/Spent Fuel Test (C/SFT), was performed for decay heat level. Also, spent fuel assembly B43, removed from the fuel temperature test (FTT) at the Engine Maintenance, Assembly, and Disassembly (EMAD) hot-cell facility in Nevada, was measured for decay heat level. The measured decay heat values of these assemblies agreed within 6 percent of the predicted ORIGEN2 values (Groff, 1980).

Results of the structural and fracture mechanics analyses of spent fuel under disposal cycle conditions showed negligible degradation of the cladding and fuel pellets from normal transportation or seismic loadings during the initial 1,000-year disposal period.

All data from the destructive examination of five fuel rods pulled from the C/SFT assemblies were collected and reviewed. These data were reduced, analyzed, and reported in a preliminary draft.

The *Interim Report on Spent Fuel Whole Rod Testing* (Einziger and Cantley, 1980) was prepared, and further testing continued until 30 September, 1980, when the program was terminated. Rods were tested at temperatures of 900, 960, and 1,060 F for 4,600, 7,700, and 1,700 hours, respectively. No cladding failures were observed.

An analysis was started to evaluate the effects of fission gas release from spent fuel disposal in geologic repositories. Worst case conditions, where up to 100 percent of the fission gas inventory would escape, were considered.

Activities During the Reporting Period

Spent Fuel Calorimetry

Documentation of the final design and performance of the spent fuel calorimetry system used at EMAD for measuring decay heat levels of Turkey Point spent fuel assemblies was completed this quarter.

Structural Analyses and Evaluation

Fuel assembly structural and fracture mechanics analyses of spent fuel under disposal cycle dynamic loading conditions were completed and documented. Results of the analyses show negligible damage to spent fuel assemblies because the resulting stresses are so low under normal expected transportation and disposal cycle conditions (e.g., 5,000 total cycles at peak stress less than 18 ksi). Fatigue crack propagation analyses, based on material property data from hydrided and irradiated Zircaloy, demonstrated that large cladding flaws are stable under the dynamic response loading conditions. All analyses were developed from models in which boundary conditions were modified to represent either fuel assembly transport or canistered configurations.

Examination of Climax/Spent Fuel Test Assemblies

Data and conclusions from the pretest destructive examination of five rods from the C/SFT assemblies were documented. Examination results include fission gas analysis, burnup and hydrogen analyses, and a metallographic evaluation of the fuel, cladding, and cladding oxides and hydrides.

The rods had been in operation for three reactor cycles with measured burnup at 31 gigawatt days per metric tons of uranium (GWd/MTU). Fission gas release (0.21 percent) was not enhanced due to the extended burnups, although gas bubble migration to grain boundaries was evident. The fuel microstructure appeared typical for fuels with this irradiation history. Gas bubble porosity was both intergranular and intragranular, with a nonuniform distribution. Radial distributions of fuel grain size indicated some grain growth toward the center with larger differences occurring at higher axial rod locations. It is apparent that most differences between metallographic samples are related to their relative axial locations on the rod. The maximum cladding temperature was calculated to occur at 113 inches from the rod bottom; this area also exhibits maximum grain size, oxide thickness, hydrogen concentration, and hydride size. Although circumferential and rod-to-rod differences were also investigated, no relationships were identified.

Whole Rod Elevated Temperature Tests

Reduction of data from the Whole Rod Test Program started this quarter. Examination of the fuel pellets has essentially been completed. Grain size, the number of fuel cracks, and the general microstructure were evaluated and then compared with pretest results. The cladding is now being analyzed; all hardness data have been graphically profiled as a function of radial distance. Relationships between hardness, oxygen content, and the photomicrographs are being investigated.

Fission Gas Release in a Repository

A perspective on fission gas release from spent fuel during geologic disposal was completed. Based on a typical proposed canister and repository design, even under the worst case of highly improbable release scenarios, fission gas pressurization presents no harmful effects to the waste form, package, or repository.

Future Activities

Each of the activities in this category will be completed. Documentation of the results will be prepared for ONWI and for public distribution under UC-70 distribution.

Spent Fuel Waste Form Assessment

Objective

The objective of this activity is to evaluate each of five candidate spent fuel waste forms for postemplacement performance, with emphasis on their ability to retard the release of radionuclides to the geology. Stabilizer filler material for the waste package is also considered in this assessment.

Progress Reported Previously

This is a new activity that started late in FY 80. No progress has been reported previously.

Activities During the Reporting Period

The purpose of this task is to assess the relative ability of four basic waste forms to contain their radionuclide inventories in a nuclear waste repository. These four waste forms are: (1) intact fuel rods in an assembly (with or without end fittings; i.e., two candidate forms); (2) vented and resealed fuel rods; (3) close-packed fuel rods (removed from assemblies); and (4) chopped fuel rods in a matrix material.

In order to reduce these four basic waste form variations to a series of well-defined cases, a matrix was developed incorporating all significant variables, including: (1) fuel rod configuration (in-grid, close-packed, or chopped); (2) stabilizer type (solid, particulate, or gas); and (3) pressure state of pin (vented or unvented).

Construction of this alternative waste form matrix produced a total of 13 physically realizable combinations of system variables. For this analysis, the chopped fuel waste form was treated as a special case of vented fuel in a solid stabilizer.

A simplified radionuclide transport model has been developed as a means of comparing the relative performance of these 13 various waste form alternatives. The model consists of a set of finite capacity radionuclide sources driving a series of time-varying transport resistances which correspond to the barrier system in the canister. The response

(radionuclide transport rate out of the canister as a function of time) for each rod will then be summed over all rods to obtain the total response of the canister. The resulting relative transport rates out of the canister as functions of time will be used to judge the relative merits of the various alternative forms.

As part of the problem definition, a set of four major tasks was developed, dealing with:

1. Fuel-leaching behavior.
2. Stabilizer performance during and after the thermal period.
3. Fuel, cladding, and system behavior during the thermal period.
4. Fuel, cladding, and system behavior after the thermal period.

These tasks were divided into subtasks primarily directed toward modeling the transport resistances of the various barriers as functions of time given material corrosion characteristics and estimates of ground-water chemistry and its rate of movement.

Future Activities

The radionuclide transport model will be exercised over various ranges to evaluate the relative merit of each of the 13 waste form alternatives. The waste form assessment study will be completed and documented by mid-February, 1981.

Spent Fuel Deterioration

Objectives

The purpose of this task is to identify properties of spent fuel which will affect the migration of radionuclides and to identify and investigate all deterioration mechanisms that might have significant impact on important aspects of the spent fuel condition during the period following emplacement but prior to major waste package breach. The range of applicability to geologic disposal and the qualification of these deterioration mechanisms are to be investigated and identified. This assessment will also consider beneficial effects of the spent fuel cladding and waste package stabilizer filler that could justify establishing design limits to preserve the former or identify desirable features that would influence selection of the latter.

Progress Reported Previously

This is a new task that started on 1 October, 1980. No previous progress has been reported.

Activities During the Reporting Period

Identification of pertinent spent fuel characteristics and deterioration mechanisms was accomplished through a series of contacts with the North American Laboratories who have conducted spent fuel leaching experiments. After an initial evaluation was made, a consensus opinion was obtained between HEDL and these laboratories concerning the need for spent fuel characterization.

This survey of the spent fuel leaching community indicated that surface area, surface structure, grain boundary condition, and fission product concentrations and positions are the most important spent fuel performance characteristics affecting the leach rate. To a lesser degree, a better knowledge of the fuel phases should be obtained. It was felt that the leaching experimenters themselves would be in the best position to look at any changes in valence states of the actinides.

The current status of characterization and the sophistication of the characterization techniques are presently not adequate. A number of preliminary experiments which better define surface area and the dynamic characteristics of the grain boundaries should be conducted prior to starting significant characterization efforts. Characterization efforts should be strongly coordinated with UO_2 and spent fuel leaching efforts, since the ultimate use of characterization is in support of spent fuel leaching studies.

It is important to define the distributions of spent fuel characteristics as a function of manufacturing variables and reactor operating conditions in order to define a typical and worst case fuel pellet. This would contribute to the establishment of a reference fuel which would facilitate the comparison of leaching results from different laboratories.

Since leaching will not occur until thousands of years after repository emplacement of the spent fuel, it is necessary to know the relative condition of the spent fuel after a long repository residence as compared with the spent fuel used for the leaching studies. Five possible mechanisms that could change the spent fuel characteristics have been identified.

1. Chemical changes in the fuel.
2. Volatile fission product migration.
3. Diffusion of gas to grain boundaries.
4. Radiation damage due to decay of fission products and actinides.
5. Fragmentation due to thermal or mechanical effects.

Future Activities

Work will continue on assessing the possible degradation mechanisms in spent fuel, and the results will be documented. Final reports will be prepared in both areas of concern in this task during this period.

Spent Fuel Characterization

Objective

The objectives of this activity are (1) to gather and correlate available information on spent fuel characteristics important to long-term resistance to release of radionuclides, (2) to develop and evaluate theoretical models or empirical correlations to describe the condition of spent fuel from the time of emplacement until breach of the waste package, and (3) to establish ranges of variability and distributions of the parameters of interest. Information produced by this task will be used to establish appropriate repository and waste package design limits and to conduct long-term performance assessments.

Progress Reported Previously

This task is a new activity that began in FY 81.

Activities During the Reporting Period

Spent Fuel Characterization Data Base

Access to the Electric Power Research Institute (EPRI) data base on LWR fuel was obtained in December, and work on compiling the data base is continuing. Work this quarter focused on gathering information for the next subtask.

Fuel Manufacturing Variables and Reactor Operating Conditions

The major effort for this subtask concentrated on data collection. Access to the EPRI data base was accomplished, and the actual work of extracting data began. Averages and ranges for several variables pertinent to this study were obtained. Seventeen fuel vendors and fuel engineers in the U.S. were polled about the importance of these parameters as they affect spent fuel characteristics at the time of discharge from reactors. In general, the responses agreed with HEDL's initial selection and assessment of these parameters.

Spent Fuel Characterization Correlations

Other than identifying those correlations known to exist or discovered in the literature searches, work has not actively started on this subtask.

Future Activities

Additional work to strengthen the results obtained for the subtask on fuel manufacturing variables and reactor operating conditions will be completed via further literature searches, more sophisticated searching of the EPRI LWR fuel data base, and further contacts with fuel vendors. Work on the spent fuel characterization data base, gathering information on the condition of spent fuel as it leaves the reactor, will receive more attention during the next few months. Information on existing empirical correlations and theoretical models will be gathered to determine the state of the art and what models must be developed.

REFERENCES

Einziger, R. E. and D. A. Cantley, 1980. *Interim Report on Spent Fuel Whole Rod Testing*, HEDL-6776, May.

Groff, A. G., 1980. *A User's Manual for the ORIGEN2 Computer Code*, ORNL/TM-7175, July.

WBS 1.2.2

Project: Spent Fuel Disassembly Alternatives Impact

Principal Investigator: E. R. Johnson Associates, Inc. (E. R. Johnson)

ONWI Project Manager: E. F. Benz

Objective

The objective of this study is to obtain an assessment of the impacts of four spent fuel disassembly alternatives on the nuclear waste isolation system relative to the disposal of unmodified spent fuel. Such impacts will include consideration of relevant technical, operational, safety, and economic factors. The disassembly alternatives will include (1) end bell removal, (2) fission gas venting and resealing, (3) fuel disassembly and close-packing of fuel pins, and (4) fuel chopping and immobilization.

Progress Reported Previously

A review was made of the design of the process and facilities for the packaging and disposal of unmodified spent fuel. This became the reference case for analyzing the impacts of the spent fuel disassembly alternatives on the nuclear waste isolation system. Preliminary reference PWR and BWR fuel designs were identified.

A review was made of the technology available for accomplishing the prospective disassembly alternatives, and the development of processes for each of these alternatives began. Identification of the additional facilities and materials necessary for accomplishing each disassembly alternative (relative to the base case) was initiated.

Preliminary packaging design parameters were developed for each alternative. The development of a methodology to compare the impacts of each alternative and the effects of human intrusion after repository closure was initiated.

Activities During the Reporting Period

Process flow sheets and descriptions were completed for end-fitting removal, fission gas venting, disassembly and pin packaging, and immobilization in a solid matrix.

Identification of the specific equipment items necessary to implement each of the four disassembly alternatives began and is in progress. The development of cost information for the equipment involved also began.

Preliminary conceptual layouts of facilities required to implement end-fitting removal, fission gas venting, and disassembly and pin packaging were completed. The development of cost information for the required processing facilities also began.

Preliminary packaging parameters were established to conform with the current Westinghouse Advanced Energy Systems Division (AESD) packaging design. These include the use of a 0.25-inch thick titanium canister inside a 1-inch thick Inconel liner which is seal-welded at the top; this results in the double containment of the spent fuel form. A particulate sand stabilizer is used within the titanium canister, and a sand filler is used between the canister and the liner. The cavity in which the liner is emplaced is backfilled with bentonite between the package assembly and the side of the cavity. The development of cost estimates on the package was initiated.

A methodology was developed for assessing the prospective exposure of both workers and the public to radiation during routine operation of the packaging facility. Operating exposures are being estimated by calculating (1) the amount of radiation that emanates from each process and support cell, and (2) the number of workers required in each operation. Maintenance exposures are being developed by estimating the extent of decontamination of equipment prior to contact maintenance, the resulting radiation levels of decontaminated equipment, the frequency of maintenance, and the number of persons involved. Prospective public exposures are being developed from estimates of both the radiation emanating from the plant and the radioactivity content of airborne and liquid releases. Radiation exposure experiences from the operation of the Nuclear Fuel Services Corporation (NFS) reprocessing plant as well as government activities are being reviewed for relevance to this evaluation.

The prospective radiation exposure of a member of a drilling crew involved in an intrusion of the repository 100 years after repository closure has been examined. Estimates of radiation-exposure levels resulting from a direct hit on a fuel package during drilling are being made, as are estimates of the probability that a drilling activity would result in an encounter with a spent fuel package.

Previously a figure of merit (FOM) methodology was developed for use in comparing the alternative disassembly methods under consideration. The criteria used to determine the FOM have been revised in an attempt to make them more independent of one another. Table 1 sets forth a description of the criteria to be used in the comparative evaluation of the disassembly alternatives.

From the work conducted to date, it appears that disassembly followed by packaging and disposal of the resulting fuel pins offers a significant advantage over the disposal of unmodified spent fuel and the other disassembly alternatives. Sufficient technology exists to design, construct, and operate a facility for performing the required disassembly operations. The waste form should be equally acceptable to that of unmodified spent fuel. Disposal of fuel pins requires smaller and fewer packages and fewer emplacement cavities in the repository than do the other alternatives. In view of the fact that the cost of the package may be high (\$25,000 to \$50,000 per package), the amount of spent fuel which can be contained in a single package will have a significant impact on waste-disposal economics. The savings expected in package costs for fuel-pin disposal, coupled with other prospective savings of this alternative, should more than offset the added costs of disassembly. A small additional increase in radiation exposure to workers can be expected, but little or no additional exposure to the public should be experienced. A higher exposure to a member of a drilling crew during postclosure intrusion can be expected if the drill strikes a canister, but there is a lower probability of such a strike occurring.

TABLE 1. CRITERIA FOR COMPARATIVE EVALUATION OF SPENT FUEL DISASSEMBLY ALTERNATIVES

Figure of Merit			
A. Technical Assessment	B. Operating Assessment	C. Safety Assessment	D. Economic Assessment
1. Status of Process Technology	1. Process Complexity	1. Operator Exposure	1. Unit Cost
(a) State of Development	(a) Number of Process Steps	2. Public Exposure	
(b) Time Required to Complete Development	(b) Amount of Material Handling	3. Accident Potential	
2. Status of Equipment Development and Demonstration	(c) Amount of Internal Recycle	(a) Criticality	
(a) State of Development	(d) Amount of Secondary Waste and Off-Gas Generation	(b) Fire	
(b) Scale of Demonstration	2. Equipment Reliability	(c) Explosion	
(c) Time Required to Complete Development and Demonstration	(a) Probability of Failure/Maintenance Requirements	(d) Natural Phenomena	
3. Amount of Prior Experience	(b) Process Latitudes	4. Occupational Safety	
4. Safeguards Effectiveness	(c) Ease of Remote Maintenance	5. Long-Term Hazard	
(a) Assessability of Special Nuclear Material (SNM)	3. Quality Assurance	(a) Migration	
(b) Dispersibility	(a) Controllability of Process	(b) Intrusion	
(c) Ease of Nuclear Material Control and Accounting	(b) Ease of Accomplishing QA		
5. Acceptability of Waste Form	4. Facility/Special Material Requirements		
	(a) Extent of Contamination of Facilities		
	(b) Extent of Use of Materials in Short Supply		

Future Activities

During the next quarter this study will be completed. Identification of equipment items and preliminary conceptual designs of required facilities will be completed. Estimates of the capital and operating costs for each disassembly alternative will be made. Assessments of the technical, operating, safety, and economic factors associated with each disassembly alternative will be made in comparison with the disposal of unmodified spent fuel. A preferred spent fuel waste form will be recommended.

WBS 1.2.3

Project: Salt-Brine-Waste-Canister Interactions

Principal Investigator: U.S. Geological Survey, Reston, VA (I-Ming Chou) and
Menlo Park, CA (M. A. Clynne)

ONWI Project Manager: J. F. Kircher

Objectives

The objectives are to determine the likely interactions among salt, brine, canisters, and waste at a range of temperatures and pressures which may be considered in disposal of high-level radioactive waste or spent fuel, and to determine the properties of the resultant reaction products at temperatures up to 300 C and pressures up to 2,600 psia.

Progress Reported Previously

Data for Brine Migration Modeling

This task began in January 1980. The Soret coefficient of NaCl is an important factor in modeling the migration rates of fluid inclusions in salt (Anthony and Cline, 1971). However, the experimental data on the Soret coefficient of NaCl are quite limited, and those in mixed electrolyte solutions do not exist. A Soret cell, designed for the direct-sampling method, has been assembled and some results obtained for 1M NaCl solution between 35 and 55 C.

In the brine migration model of Anthony and Cline (1971), the salt concentrations are in molar units, while most of the literature data for salt solubilities are in molal units or in weight percent. Density data for brines are needed to convert the concentration units. Densities of halite-saturated WIPP-A and NBT-6 brines have been calculated from 0 to 100 C using the Potter-Haas (1978) model. These calculated densities are within 1 percent of those measured. In order to further verify or modify the Potter-Haas model, the densities of various binary solutions as well as brines will be measured using a calculating precision density meter. The density meter has been set up and calibrated with 1 to 6M NaCl solutions.

Solubility of Strontium and Cesium in Complex Brines

This task began in January 1980. The ^{90}Sr and ^{137}Cs isotopes are the major heat sources in fission products in spent fuel aged more than five years. Brines that might come into contact with a fuel assembly would tend to dissolve strontium and cesium in the assembly as chlorides. Such a fluid contains its own energy source and could migrate and change the thermal evolution of the repository from designed values. In order to better assess the problem, a knowledge of the solubility of strontium and cesium in complex brines is necessary. The solubility relation in the system $\text{SrCl}_2\text{-NaCl-H}_2\text{O}$ has been determined from 18 to 115 C, and the solubility of SrCl_2 in complex brines has been summarized.

Activities During the Reporting Period

Data for Migration Modeling

The Soret coefficient of NaCl in 3 M NaCl solution between 30 and 60 C has been determined. The results indicate that, in this temperature range, the Soret coefficients of NaCl are essentially the same as those for 1 M NaCl solution. Measurements for 6 M NaCl solution have been initiated. It was found that the gravimetric method for determining the amount of sample solution withdrawn from the Soret cell was not adequate for these concentrated solutions. A new volumetric technique for sampling and analyzing solutions has been developed. This technique is efficient and gives reproducible results.

Densities for the binary solutions of NaCl (1.0 to 6.1 M), KCl (0.5 to 4.5 M), MgCl_2 (0.5 to 3.0 M), and CaCl_2 (0.5 to 5 M) have been measured at 25 C. The concentrations for the CaCl_2 and MgCl_2 solutions were checked by atomic absorption analysis. The measured densities agree very well with the published values. The PVTX system was calibrated using the volumetric properties of steam. Standard procedures have been established for calibrating the system and for determining the volume of the pressure vessel.

Solubility of Strontium and Cesium in Complex Brines

A constant temperature bath system has been set up to study the solubility relation in the system $\text{CsCl-NaCl-H}_2\text{O}$ at 1 atm and between 18 and 120 C. A standard procedure for sample preparation as well as solubility measurement has been established. The data of Plyushchev et al (1957) are the only data available for this system, and they are not internally consistent. The experimental approach has much better compositional control and will provide higher quality results, as indicated by the preliminary data.

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WBS 1.2.3

Project: Desiccant Materials Screening

Principal Investigator: Lehigh University (D. R. Simpson)

ONWI Project Manager: D. P. Moak

Objective

The objectives are to determine the phase relations in the Ca_2SiO_4 – CaNaPO_4 system and to determine the suitability of selected compositions for backfill by hydration reactions.

Progress Reported Previously

Using quenching experiments, the phase relations of the join CaNaPO_4 – Ca_2SiO_4 were studied. Above 1430 C there is nearly complete solid solution between the end members, thus negating the existence of an intermediate compound. These high-temperature phases are easily quenched. At lower temperatures, a large miscibility gap and polymorphic relations were found. These findings refute the intermediate compound hypothesis, although for intermediate compositions the high-temperature compounds are extremely persistent to lower temperatures. Such high-temperature intermediate compounds are the stock material for hydration and sorption studies.

Activities During the Reporting Period

The generalized relations of the binary CaNaPO_4 – Ca_2SiO_4 at temperatures above 1160 C were reported. At lower temperatures, the reactions were sluggish and required long-duration experiments. A series of determinations made at 888 C had results consistent with the relations shown in Figure 1. Experiments are in progress at 950 C and 800 C.

Four products of synthesis above 1380 C were used to start hydration studies. The compositions are:

1. CaNaPO_4
2. $(\text{CaNaPO}_4)_{50} (\text{Ca}_2\text{SiO}_4)_{50}$
3. $(\text{CaNaPO}_4)_{20} (\text{Ca}_2\text{SiO}_4)_{80}$
4. Ca_2SiO_4

Each powdered product is supported in a glass tube by a fritted glass filter. Water was then drained through the powders, and the filtrate was collected. After this initial washing, a daily aliquot of solution was added to each filter tube containing the powdered samples and the filtrate was again collected. Because of the fine-grained nature of the powders, the solution was held in the sample by capillary action until displaced by the next day's aliquot of solution. Thus, the rate of filtration was slow.

FIGURE 1. GENERALIZED RELATION OF THE BINARY $\text{CaNaPO}_4 - \text{Ca}_2\text{SiO}_4$

The numbers indicate the mean index of refraction.

Three series of experiments are in progress for each of the four samples. For the first series, the solutions contain 0.01 M NaI, the solution for the second series contains 0.01 M CsCl, and the third solution contains 0.01 M SrCl_2 . Rhenium is being procured for a fourth series. At this stage, rhenium is being used as a proxy for technetium because of the similarity in their chemical properties.

The filtration of the solution through the four solid samples provides data for two separate processes. First, information is gained on the rate of hydration by the change in permeability of the sample. Second, by analysis of the filtrate, and then by analysis of strata in the filter bed, the sorptive properties and the reactions in the filter bed will be established.

At this stage, several of the series are nearing completion because of the reduction in permeability of the filter bed. These experiments are at 20 C.

Products of the hydration are likely to be apatite, a calcium phosphate, and tobermorite or other hydrous calcium silicates. These hydrous products may sorb ions as rhenium, strontium, cesium, and iodine by surface processes, or they may incorporate the ions in their structure by a solid-solution process. The retention of the foreign ions will be a function of the manner of incorporation in or on the crystals. For this reason, hydration experiments are in progress to establish the manner of uptake of the foreign ions. Anhydrous products from the high-temperature synthesis were sealed with a foreign ion-bearing solution in gold or Teflon. The sealed charge was then subjected to elevated temperatures at sufficient confining pressure to maintain the integrity of the sealed charge. The temperatures for these experiments were 150 C and about 320 C.

Anhydrous CaNaPO_4 hydrates under the hydrothermal condition to form apatite, and the residual solution is strongly basic. This apatite has lattice parameters consistent with published values for the pure end member. However, CaNaPO_4 hydrated in the presence of 0.01 M CsCl had enlarged a and c axes, an indication that there is some substitution of cesium in the structure. The lattice parameters of apatite synthesized in the presence of 0.01 M NaI are within the range of error when compared to hydroxylapatite. Synthesis with increased concentration of foreign ions in the solution is continuing.

Other hydrations under hydrothermal conditions involved intermediate compositions. With hydration, the phosphate in the material form is either apatite $[\text{Ca}_5(\text{PO}_4)_3(\text{OH})]$ or monetite (CaHPO_4) . The calcium or calcium sodium silicates which form in this system by a hydration reaction are presently unidentified. The hydrated calcium silicates formed are poorly crystallized and consequently may have desirable sorptive properties.

Anhydrous $(\text{CaNaPO}_4)_{50} (\text{Ca}_2\text{SiO}_4)_{50}$ was held under water for several months at 20 C. The hydration product is poorly crystallized and the reaction is incomplete. This partially reacted material was analyzed by thermal gravimetric analysis. The sample showed weight loss starting at 63 C. This is certainly sorbed water. At 160 C, there is a second weight loss which must correspond to structural water. These data will contribute to the identification of the hydration product.

It has been established that high-temperature α -type compounds can be easily quenched from intermediate compositions in the binary $\text{CaNaPO}_4\text{--Ca}_2\text{SiO}_4$. The high-temperature compounds hydrate. Some hydration products incorporate foreign ions, have cementing properties, and have crystallinity, suggesting that they have desirable sorptive properties. The exact manner of the incorporation or sorption is being determined.

Future Activities

Most of the effort for the next quarter will center on the hydration reaction. The manner of incorporation of the foreign ions in the hydration products will be established by analyses of the filtrates and strata in the filter beds. At this stage, the analytical approach and apparatus appear adequate. If the hydrated filter bed is suitable, physical properties of the bed will be determined.

WBS 1.2.3

Project: Development of Backfill Materials

Principal Investigator: Battelle Pacific Northwest Laboratories (E. J. Wheelwright)

ONWI Project Manager: D. P. Moak

Objectives

The objectives of this project are to define the optimum attributes of backfill materials and to develop the single-component or multicomponent backfill material that most nearly satisfies the prescribed attributes.

Progress Reported Previously

Backfill requirements were defined and ordered by priority. Potential backfill materials were screened and, where possible, ranked in terms of attributes. These included bentonite-quartz mixtures, powdered metals and graphite, and synthetic and natural zeolites. These materials were evaluated in water and radionuclide migration experiments, by physical property measurements, and as to chemical stability.

Activities During the Reporting Period**Summary**

The major activities during this reporting period have been: (1) the acquisition of experimental quantities of candidate backfill materials, (2) continuation of radionuclide distribution coefficients with the materials, (3) initiation of compaction tests, and (4) design and construction of the permeability cell systems.

Backfill Material Evaluations

Science Applications, Inc. assisted PNL during FY 80 in (1) defining and ordering by priority the desired attributes of the backfill materials, and (2) determining what quantitative data exists in the open literature on possible candidate backfill materials. A candidate list of backfill components has been developed from this information and is shown in Table 1. Using these individual materials and homogeneous mixtures of two or more components, a program of laboratory testing has been initiated to test three ground-water compositions and selected radioisotopes (Barney and Wood, 1980).

Batch distribution (K_d) values have been determined using bentonite clays (Volclay® and Accofloc-350®), several synthetic zeolites, glauconite (ferrosand), and iron metal powder as examples of commercially available materials for backfill evaluation. Samples of each were contacted with a low-ionic-strength ground water (pH 8) containing ^{125}I , ^{85}Sr , ^{137}Cs , or ^{99}Tc . The samples were equilibrated for 7 days, the water fraction filtered with a 16 A membrane filter

TABLE 1. SELECTED BACKFILL MATERIALS

-
-
- 10 to 30% Sodium Bentonite in Quartz Sand
 - Hydrophobic bentonite
 - Bentone®
 - Cab-O-Sil®
 - Natural and Synthetic Zeolites
 - Clinoptilolite
 - 13X
 - AW-400®
 - Zeolon-900®
 - Ferrosand
 - Metal Powders or Fibers
 - Iron
 - Aluminum
 - Titanium
 - Graphite
 - Lead Oxide
 - Mineral Eh-pH Buffers
 - Hematite $\rightleftharpoons$ Magnetite
 - Magnetite $\rightleftharpoons$ Siderite
-
-

cone, and the resulting samples counted for radionuclide content. The Kd results show that cesium and strontium are retained by the clays ($2 - 5 \times 10^3$) and zeolites ($10^3 - 10^6$). Iodine, as expected, was not retained by any of the materials tested. Technetium was retained by the iron metal powder (10^3). Additional tests have been initiated to obtain batch distribution values for tuff, basalt, and salt brine ground waters and to determine the stability of various tracers in the above waters as a precursor to initiation of the dynamic flow-through system. The results are shown in Table 2.

Backfill densities are being measured, as a function of pressing conditions, for individual clays and on compacts of homogeneous mixtures of two or more materials. An air-hydraulic press is being used to obtain 1 to 5-cm-diameter test pellets. Powdered zeolites, quartz sand, metal powders, and clays are being mixed in various ratios for this test program. Compaction of as-received Volclay using 2, 6, 10, 14, and 18 tons/inch² has resulted in compaction densities

TABLE 2. BATCH DISTRIBUTION VALUES FOR
BACKFILL CANDIDATES

	Distribution (Kd), ml/gram ^(a)				Final pH
	Cs	Sr	I	Tc	
Volclay (smectite)	5.2E3	2.0E3	6.3	14.5	7.96
Zeolon-900 (mordenite)	> 3.4E5	> 1.6E6	< 0.3	4	8.51
13X (faujasite)	7.8E3	> 1.6E6	1.4	6	8.52
Ferrosand (glauconite)	7.0E4	2.1E3	4.9	15	7.08
Iron (metal powder)	8	1.1E2	7.6	2.5E3	9.96
Accofloc-350 (Na bentonite)	2.4E3	3.6E3	< 0.4	10	8.37
Zeolite 4A (Type A)	8.5E3	8.6E3	< 0.4	< 0.4	9.52
AW500 (erionite)	1.9E4	4.4E3	< 0.4	19	8.28
Blank	—	—	—	—	7.84

$$(a) \quad Kd = \frac{\text{Standard} - \text{Aqueous}}{\text{Aqueous}} - \frac{\text{Sample Volume (20 ml)}}{\text{Weight of Solid (0.5 gram)}} = \text{ml/gram.}$$

Standard = filtered blank or filtered acid sample.
Aqueous = filtered aqueous count rate.

of 1.92, 2.08, 2.14, 2.16, and 2.20 grams/cm³, respectively. Physical measurements of the individual clays and compacts (e.g., thermal conductivity as a function of density) will be initiated.

Constant-Volume Permeameter-Cell System

The system to measure water and radionuclide migration through candidate backfill materials has been designed, and the components are being fabricated and assembled. Calculations based on Darcy's law for flow in porous media indicate that a pressure of about 34 MPa (5,000 psi) will be needed to force 1 ml/day of water through a 5-cm-diameter x 1-cm-thick disk of compacted bentonite clay (~ 2 grams/cc density). The system has been designed to withstand these high pressures. Figure 1 shows the components of this system.

Water will be forced through the system by a high-pressure liquid chromatography pump capable of flows of 0.01 ml/min at pressures of 42 MPa. A back-pressure relief valve will actually control the pressure within the system and will vent excess flow back to the reservoir.

FIGURE 1. SCHEMATIC OF WATER-RADIONUCLIDE MIGRATION SYSTEM

FIGURE 2. SCHEMATIC OF PERMEABILITY CELL

System pressure will be measured by a transducer ahead of the permeability cell. For very impermeable materials, flow rates will be measured by collecting water that has passed through the backfill in a capillary tube. Larger volume collectors will be needed for more permeable backfill candidates.

Figure 2 shows the permeameter cell itself. Water enters through the bottom piston and is distributed on the backfill sample face by a porous metal disk. The sample itself will be 5 cm in diameter, but the height can be varied between 0.5 and 5 cm. After passing through the sample, the water is recollected and flows out to the capillary tube. A thermocouple near the top of the sample will monitor the temperature of the test, and a load cell will measure the swelling pressure developed by backfill materials as they saturate with water.

All wetted parts of the system will be of stainless steel, sapphire (pump pistons), Kel-F®, and Teflon®. Tests under anoxic conditions should require little, if any, modification to the system.

REFERENCE

Barney, G., and B. J. Wood, 1980. *Identification of Key Radionuclides in a Nuclear Waste Repository in Basalt*, RHO-BWI-ST-9, Rockwell Hanford Operations, Richland, WA.

WBS 1.2.3

Project: The Chemistry and Physics of Nuclear Waste-Rock Interactions

Principal Investigator: Pennsylvania State University (W. B. White)

ONWI Project Manager: D. P. Moak

Nuclear Waste-Argillaceous Rock Interactions**Objective**

The objectives are (1) to repeat certain key waste-rock interaction experiments using more reactive younger shales and determine if there is a change in the expected reaction chemistry; (2) to continue the canister-waste-rock interactions and the sequence of experiments on the transport mechanisms of radionuclides in massive shales to determine the distribution of elements along the diffusion gradient and the partitioning of elements among the mineral phases to provide a basis for deducing the transport mechanisms in the actual repository environment; (3) to synthesize the rare earth reaction product phases and determine their X-ray powder diffraction patterns, chemical composition, and the relationships of the various structures as a function of a rare earth ion.

Progress Reported Previously

Progress on the work begun on 1 October 1977 relates to the identification of chemical reactions and transport mechanisms for reactions of prototype nuclear wastes with argillaceous rocks under specified temperature, pressure, and fluid composition conditions. The initial experiments dealt with the stability of the waste forms under hydrothermal conditions, especially in the presence of bittern brines. The results, glass-brine interactions, ceramic-brine interactions, and spent fuel-brine interactions are complete and will appear as a series of topical reports.

The second sequence of experiments dealt with binary interactions between simulated waste forms with generic shale rocks in the presence of the specified fluids. Resulting from this is the identification of some 30 compounds as the reaction products of the wastes with the aluminosilicate and other minerals found in the shales. Reaction runs of varying times determined the rate at which steady state is attained and whether the initial reaction product assemblages are modified after longer reaction times.

The third sequence of experiments included the canister as an interaction component, and the fourth is incorporating transport processes as well as bulk chemistry. The behavior of specific radionuclides and their reaction products has been investigated. Cesium has been studied extensively, including its interactions directly with the various shales and their component clay minerals taken separately, as well as the stability of the principal cesium reaction product phase, pollucite, in the presence of the various fluids. In a similar manner, the stability relations of the alkali uranyl silicates, which often appear as reaction product phases, were determined, and compatibility triangles for the system $K_2O-UO_3-SiO_2-H_2O$ at 200 and 300 C were constructed.

Activities During the Reporting Period

Continued examination of long-duration experiments yielded no new phases or generalizations beyond those already noted. No further direct effort on the problem of oxygen buffers has been undertaken. Two items bearing on this question should be noted. The Teflon capsules appear to be permeable to hydrogen, and an external oxygen buffer seems to work; thus, the iron-uranium-oxygen experiment should be feasible using an inner capsule of Teflon and an outer capsule of gold or silver.

Some capsules from the shale-waste-metal-fluid set of experiments have been opened and analyzed for both solution chemistry and solid characterization. The capsules opened were all run at 250 C with PNL 76-68 glass. X-ray diffraction (XRD) analysis of the shale plus waste mixture produced a weak pattern, with quartz being the only recognizable phase and, apparently, with a substantially reduced abundance compared to the starting material. Scanning electron microscope (SEM) examination of the shale plus waste revealed the presence of analcime-pollucite solid solution crystals and what is thought to be a newly formed sheet silicate. Cesium was present at or below the detection limit ($10 \mu\text{g/ml}$). In a few capsules, calcite crystals were recognized, and in all cases, a significant amount of strontium was detected in the calcite. Strontium was also present at or below the detection limit for the fluids in contact with these experimental runs. Uranium was present as uranophane and/or weeksite only in the capsules with copper or stainless steel, but it was not recognized among the solid phases in the runs with mild steel. A tentative identification of uraninite was made in the capsules with Zircaloy. The Zircaloy strip was discolored after the run; that is, it was a very dark brown or black. SEM and energy dispersive X-ray analyses of the surface of the strip identified uranium in addition to the zirconium. X-ray diffraction (XRD) analysis done on a piece of the strip produced the pattern for zirconium, plus a very weak and broad peak at the correct location for the most intense peak of uraninite (this would also be consistent with zirconium oxide). Mild steel strips were severely rusted, leaving a core of metal in a thick rind of "rust". Copper strips were noticeably corroded, but not grossly reduced in size. Stainless steel was not visibly attacked.

Presence of mild steel or Zircaloy in the shale-waste-metal-fluid experiments appears to have inhibited formation of weeksite or uranophane from PNL 76-68 glass. The interpretation offered is that the mild steel and Zircaloy reaction with the mineralizing fluid produced reducing conditions and uranium was present as the plus-four species (either as uraninite or presumably absorbed on the rust). Stainless steel did not react, hence there was no mechanism for reducing the uranium. The reaction of copper apparently reduced the oxygen fugacity to the copper-copper oxide buffer which appears to be above the stability field of uraninite and within the stability fields of uranophane and weeksite.

Binary interactions between the waste elements and clays or zeolites or shales in the presence of water are being investigated under hydrothermal conditions (Table 1). The overall set of binary interactions investigated encompasses 80 experimental runs.

A new set of sample compositions was prepared to continue the study of the phase stability of alkali uranyl silicates. Samples were run under hydrothermal conditions and analyzed. The results are consistent with earlier runs. The left side of the phase diagram remains unchanged (Figure 1). The most recent data have resulted in the addition of four new compatibility triangles. However, the new data did not yield sufficient information to allow the phase relations in the lower right (near the UO_3 apex) to be determined. Irradiation (IR)

TABLE 1. EXPERIMENTAL MATRIX

Antrim Shale Brallier Shale Conasauga Shale Salona Shale Chlorite Illite Kaolinite Montmorillonite Clinoptilolite Mordenite	X	$\left[\begin{array}{l} \text{Sr as Sr(OH)}_2 \\ \text{Sr as SrCl}_2 \\ \text{La as LaCl}_3 \\ \text{Nd as NdCl}_3 \end{array} \right]$	X	$\left[\begin{array}{l} 200 \text{ C} \\ 300 \text{ C} \end{array} \right]$
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The shaded region indicates the area of the phase diagram where it has not yet been possible to determine the phase relations.

FIGURE 1. TENTATIVE PHASE FIELDS PROPOSED FOR THE SYSTEM $\text{K}_2\text{O}-\text{UO}_3-\text{SiO}_2-\text{H}_2\text{O}$

analyses were completed for each sample using the KBr pellet technique. In addition to confirming the XRD analysis, the IR spectra yielded "fingerprints" for most of the commonly occurring phases in the system $\text{K}_2\text{O}-\text{UO}_3-\text{SiO}_2-\text{H}_2\text{O}$. The IR spectrum of weeksite indicates that the water molecules in its structure occupy definite crystallographic sites rather than being randomly distributed. These spectra will be useful in further studies of this system as well as in other waste-rock interaction systems.

Future Activities

Analysis of long-duration and shale-waste-metal-fluid experiments will continue. The first round of mineral synthesis will be characterized. Preparations will begin for the trace element-spiked mineral synthesis. Analyses of solutions for strontium, lanthanum, and neodymium and XRD characterization of solid reaction products from the 300 C/300 bars/12-week interaction runs of clays, shales, and zeolites with Sr, La, and Nd will be undertaken. Work will continue on the $\text{K}_2\text{O}-\text{UO}_3-\text{SiO}_2$ phase diagram. Experiments on ion migration will be initiated.

Repository Behavior Deduced from Literature Data

Objective

The objective is to make a systematic survey of the relevant geologic literature, summarize this information in a way useful for repository design, and draw such conclusions as possible about the repository processes from the experimental data.

Activities During the Reporting Period

The review of the geologic literature in the attempt of anticipating near-field repository behavior has focused upon two major aspects. The first is the construction and application of various petrogenetic models to the problem of near-field reactions. A petrogenetic model attempts to predict stable mineral phases under a wide range of P-T-X conditions, with emphasis on defining trends in mineral stability under changing environmental conditions. The second is the consideration of the major environmental factors that influence the stability of various potential radionuclide-bearing zeolite minerals.

A major step in using petrogenetic grids as a tool for predicting mineral assemblages to define what is the equilibrium mineral assemblage at the P-T-X conditions being considered. There are rigorous definitions preferred by chemists where equilibrium represents the minimum free energy of a system and where reactions are assumed to proceed rapidly to completion. However, reactions involving mineral phases are seldom instantaneous, and local minima in the free-energy surface are common; thus metastable phases are produced that

appear to be stable under existing conditions but do not represent minimum free-energy values. The consideration of reaction kinetics has usually been ignored in predicting near-field reactions, both in the rate of formation of "equilibrium" phases and the rate of dissolution of these phases when environmental conditions change. The absence of thought in this direction is most likely due to the limited information in the geologic literature on reaction rates among common low-grade metamorphic minerals, e.g., zeolites. Therefore, any conclusions concerning the rates of mineral reactions, toward and away from stable equilibrium phases, will need to be determined from available thermochemical data and theoretical considerations.

The second aspect poses different problems. Instead of the paucity of information available to solve various diagenetic models, there is a considerable amount of information in the geologic literature on factors that influence zeolite mineral stability. The major problem is that much of the information is spotty or difficult to compare with other studies, while some results directly contradict others. The task, then, is to sift through all the information and arrive at several generalizations that can be incorporated into simplified petrogenetic models for predicting near-field reactions.

Some of the apparent contradictions include closed-system experiments where the presence of very low P_{CO_2} (>0.007 percent) inhibits zeolite mineral reactions, while in nature zeolites are often found associated with various carbonate minerals. A major thrust of this task is to deduce generalizations that explain the lower equilibrium temperatures of zeolite mineral reactions in nature compared to experimental results. Factors such as decreased fluid pressure/total pressure ratios, decreased water activity, and increased time for reaction all result in these lower natural equilibrium temperatures. Attempts are being made to cross-correlate these factors into one simple model.

Along with the investigation of zeolite stability relationships, a preliminary search of the literature of clay mineral alteration under diagenetic, low-grade metamorphic conditions has been initiated. The underlying assumptions of studying these particular major mineral groups are that they would be the most common products of hydrothermal alteration of virtually any proposed host rocks and also the most useful in retarding radionuclide migration.

The chemical and mineral composition of the host rock and of the associated interstitial fluids play a major role in the stability relationships of zeolite and clay minerals. There is relatively good information, from thermodynamic and field analyses on the occurrences of various zeolite minerals in differing water compositions; this can be visualized on a series of activity-activity stability field diagrams. The influence of host rock composition has proven more difficult to define; problems abound in the chemical similarity of many host rocks that produce vastly different zeolite minerals and also the large variation in composition of presumably similar host rock, e.g., shales. An insufficient number of chemical analyses of host rock can be directly associated with the secondary minerals, thereby hindering any mass balance considerations.

Future Activities

The focus of the investigation of the geologic literature on repository behavior will shift toward the writing and synthesizing of the collected information. The primary intention is to generate a series of models to explain the mechanisms that influence the common diagenetic reactions associated with each major repository rock type.

Brine Migration in Salt

Objective

The objectives are (1) to finalize the characterization of Avery Island salt to determine what new analytical procedures are necessary to differentiate between samples for the migration experiments and (2) to learn more about thermal gradient migration of brine inclusions in polycrystalline salt with the use of the brine migration cell and alternative tracer techniques.

Progress Reported Previously

Methods of preparing hot-pressed samples of NaCl with carefully controlled microstructures were described previously. These synthetic materials have been used in migration experiments. The design and construction of two types of migration cells have been described. Previous results using synthetic salt showed that large thermal gradients on the order of 50 degrees/cm produce migration rates with orders of magnitude greater than predicted by theory. The migration paths seem to be relatively independent of salt microstructure. When smaller thermal gradients of 3 to 7 degrees/cm were used, the migration velocity in the synthetic salts was 10 times higher than the calculated values of 4 to 5 cm/year. From microstructural examination it was evident that the migration paths were principally along grain boundaries.

Future Activities

The migration cell experiments now in progress should produce results during the next quarter. Preliminary results on the feasibility of the tracer techniques are expected along with some hot-stage evidence of the effect of grain boundaries on brine migration.

WBS 1.2.3

Project: High-Level Waste Interactions

Principal Investigator: Sandia National Laboratories (M. A. Molecke)

ONWI Project Manager: J. F. Kircher

Objective

The objectives are:

1. To quantify the corrosion and metallurgical behavior of candidate HLW canister or overpack materials under environmental conditions to be expected in a salt repository and overtest conditions thereof. To provide and verify predictive models or assessments of the long-term barrier effectiveness of HLW canisters and overpacks based on laboratory, hot-cell, field, and in situ experimental results.
2. To develop and test backfill barriers to retard ground-water intrusion toward the waste package and to significantly delay radionuclide migration (via "getter" additives). This backfill getter is to be compatible for HLW or TRU waste isolation in a salt repository.

Progress Reported Previously

Sandia has been experimentally involved in the corrosion behavior of HLW and TRU waste container materials since FY 76, and in studies for backfill-getter development since FY 77. Assessment of these near-field interactions was to initially support consequence assessments being conducted for the WIPP environmental impact statement and subsequent public hearings. Current studies, since July, 1980, have been conducted in a generic manner applicable to the isolation of wastes in a salt (bedded or domed) repository.

Several candidate canister-overpack alloys have been selected after multiple screening tests and assessments for superior corrosion resistance in brines. Several titanium alloys have received additional detailed testing.

Initial screening studies on candidate getter materials have been completed. Advanced studies of chemistry and materials testing, geoengineering, and effectiveness modeling are in progress.

Activities During the Reporting Period**Canister-Overpack Development**

Stress corrosion testing of TiCode-12 sheet and rod samples has been extended to more extreme environments in an attempt to identify conditions in which the material shows macroscopic evidence of stress corrosion cracking. Samples have included electron beam (EB) and tungsten-inert gas (TIG) weldments. Tests have been performed at pH values of 1 to 6 and

temperatures of 25 to 200 C. In some cases, a refreshed autoclave system was used in order to evaluate the effect of dissolved oxygen. All tests during this period were performed in seawater, since the oxygen concentration in seawater exceeds that in WIPP brines. Work has also begun on heat-tested TiCode 12 samples to screen for possible sensitization effects.

Initial screening experiments by slow strain rate testing have been conducted on mill-annealed TiCode-12. These experiments in hot chloride solutions have shown some indication of environmental interactions at strain rates of 10^{-6} sec $^{-1}$, where analysis of the fracture surfaces showed limited areas of cleavage, a failure mode usually associated with hydrogen embrittlement. Subsequent experiments at rates of 10^{-7} sec $^{-1}$ are under way to assist in determining if environmental interactions are occurring.

A successful overpack material must possess an appropriate combination of mechanical properties, corrosion resistance (both general and localized), and resistance to environmentally assisted cracking (hydrogen embrittlement or stress corrosion cracking). All of these properties of importance are known to be affected by the microstructure; furthermore, microstructures can be altered by processing. There are indications, for example, that one of the candidate alloys, Ti-50A, is "sensitized" by some heat treatments, and that such sensitized material has inferior corrosion resistance.

This work is designed to enable processing of the TiCode-12 and Ti-50A alloys to provide the best combination of desired properties and to ensure that fabrication methods such as welding do not substantially degrade the materials. The approach being used is to establish time-temperature-transformation (TTT) diagrams, and then to determine the general corrosion resistance and environmental cracking behavior of materials whose microstructures lie in various areas of the TTT diagrams.

Plates of both titanium alloys, 0.62-cm-thick, have been procured and cut into sections suitable for heat treatment. Heat treatments have been on TiCode-12; a total of 183 material conditions will be evaluated. A 16-month contract, to begin in mid-January, 1981, has been placed with a subcontractor, Titanium Metals Corporation of America, for assistance in this area. This work is to ensure that TiCode-12 can be produced in the thicknesses required for overpacks to be used in a salt geologic repository (approximately 1 to 2 mm) with appropriate properties. Since TiCode-12 was developed as a heat exchanger tube material, only thin sheet has been required, and process controls necessary for production of thicker sections (e.g., 6 mm or more, if desired) have not been previously developed.

Studies to elucidate the mechanisms of corrosion on TiCode-12 are in progress. Auger electron spectroscopy (AES) and electrochemical techniques are being used. The goal of this effort is to develop an analytic, predictive model on the corrosion of TiCode-12 in a salt geologic repository environment. This is made difficult by the fact that no localized attack (e.g., pitting, cracking) has been observed so far on TiCode-12.

The PNL-Sandia joint HLW total-system test was initiated in October. The first phase of this experiment using uranium-doped 76-68 glass (rather than fully loaded, radioactive HLW glass) was started on 25 October. This test will continue until 26 January 1981. Test conditions include a temperature of 250 C and a 2,000 psi overpressure of argon. Brine solutions were withdrawn (and filtered) hot from the autoclave after 30 and 52 days of operation. Liquid analyses have begun.

Discussions and preliminary planning of a joint Sandia-German in-mine corrosion test of a TiCode-12 overpack in the Asse salt facility has taken place this quarter. Discussions with several members of the Institut für Tieflagerung (Braunschweig, FRG) were conducted in September, October, and December. This test is scheduled to begin in approximately September, 1981, at the Asse facility. It involves emplacing a 2-meter-long by about 0.3-meter-diameter TiCode-12 overpack, surrounding a steel heater (HLW canister simulant), into a borehole in the floor of a salt drift; the mine room will be flooded with saturated brine during the test. Overpack and heater fabrication will be provided by Sandia. The initial test design will be a joint Institut für Tieflagerung and Sandia effort, as will be the post-test analysis.

A technical presentation on the Sandia HLW canister-overpack program and results and also the backfill-getter development studies was given to the National Academy of Sciences—Waste Isolation Systems Panel in Washington, DC, on 2 December 1980.

Technical interchanges between the canister-overpack corrosion and metallurgy program at Sandia and the Kernforschungszentrum Karlsruhe (FRG) were accomplished at Sandia in October and at Karlsruhe in December.

Backfill-Getter Development

The paper by Nowak (1980) presents the progress in the backfill-getter studies. Some of the results include: Pu(IV) was found to have empirical, batch Kd values, on bentonite and bentonite-sand backfill mixtures, in WIPP brines A and B, of approximately 520 to 36,000 ml/gram, depending on conditions. Similarly, Am(III) batch Kd values ranged from 9,000 to 9,500 ml/gram.

A tailored mixture of bentonite, charcoal, synthetic mordenite, and sodium titanate could be an effective backfill material. Estimates of backfill effectiveness in terms of radionuclide retention times (times to breakthrough at 1 percent of entering concentration) can be calculated for retarded diffusion through bentonite- and getter-containing backfills. For example, as shown in Table 1, a 3-foot-thick backfill composed of 30 weight percent bentonite and 70 weight percent charcoal could retain plutonium, americium, and TcO_4^- for more than 10^4 years and I^- for more than 10^3 years in brine A. Designed for retention of cesium and strontium as well, a backfill composed of 30 weight percent bentonite, 40 weight percent synthetic mordenite, 10 weight percent sodium titanate, and 20 weight percent charcoal could retain cesium for approximately 600 years, strontium for approximately 700 years, plutonium and americium for more than 10^4 years as above, TcO_4^- for approximately 4,000 years, and I^- for approximately 400 years. These retention times can be significant contributions to the radionuclide isolation capability of a waste package system.

Experimental techniques for making direct migration measurements in solid backfills, as opposed to batch Kd measurements, have been developed and implemented. Results will be given in the next quarterly report.

TABLE 1. CALCULATED RETENTION TIMES FOR A THREE-FOOT-THICK BACKFILL LAYER, ASSUMING RETARDATION BY LINEAR SORPTION FROM BRINE A

Backfill Composition, percent	Retention Time, years	
30 bentonite 70 charcoal	Pu: 5×10^4 Am: 2×10^4	TcO ₄ ⁻ : 1×10^4 I ⁻ : 1×10^3
30 bentonite 40 mordenite 20 charcoal 10 sodium titanate	Pu: 5×10^4 Am: 2×10^4 Cs: 6×10^2 Sr: 7×10^2	TcO ₄ ⁻ : 4×10^3 I ⁻ : 4×10^2

Bulk density = 2 grams/cm³.
 Effective porosity = 0.1.
 Migration is by molecular diffusion, and breakthrough is defined at 1 percent of entering concentration.

Future Activities

Studies initiated this quarter will be continuing with an expanded effort. Milestones planned for the second quarter of FY 81 are as follows:

- Shutdown of first cold-test of PNL-Sandia joint HLW total-system test; begin post-test analyses.
- Initiate consultation on practicality assessments of backfill fabricability and emplacement.

REFERENCE

Nowak, E. J., 1980. *Complete Backfill Materials for the Radioactive Waste Isolation by Creep Burial in Salt*, SAND80-1377C, Sandia National Laboratories, Albuquerque, NM.

WBS 1.2.3**Project:** Container Barrier Development**Principal Investigator:** Pacific Northwest Laboratory (R. E. Westerman)**ONWI Project Manager:** D. P. Moak**Objective**

The objective of this project is to develop materials for the structural engineered barrier components of high-level nuclear waste packages. Structural barriers include the canister, the overpack, and the hole sleeve. The barrier elements must contribute substantially to a barrier system that will hold radioisotope release rates to near-zero values for time periods of ~1000 years.

Progress Reported Previously

This project has been under way at PNL in essentially the same form, and with the same basic objectives, since January, 1979. It is now sponsored by ONWI. Until the current fiscal year, however, it was funded by the Division of Waste Products and was reported on as part of the High-Level Waste Immobilization Program. A document by Westerman (1980) describes the project results prior to October, 1980.

Activities During the Reporting Period**Introduction**

Candidate barrier element materials are being evaluated by means of corrosion screening tests, using Hanford ground water at 250 C. The most promising candidates will be subjected to in-depth enviromechanical testing.

A dynamic autoclave facility is being constructed to expose specimens simultaneously to gamma radiation and high-temperature corrosive environments. A slow-strain-rate test facility also is being constructed to determine the susceptibility of metals to embrittlement in repository environments.

Approach

Selecting and evaluating materials for long-lived waste package barrier elements involve (1) anticipating material failure modes (e.g., general corrosion, pitting, stress corrosion cracking); (2) screening a large number of candidate materials in relatively inexpensive tests; and (3) rigorous in-depth testing of the best performers under the conditions of temperature, pressure, water chemistry, and radiation closely simulating those expected in the repository.

Corrosion screening tests are currently under way. Construction of a facility to permit exposure of specimens simultaneously to a high-temperature corrosive environment and to gamma radiation is in progress. An apparatus for testing slow strain rate is also under construction.

Screening Tests

Some iron-, nickel-, copper-, titanium-, and zirconium-base alloys are undergoing corrosion screening tests in a simulated Hanford (basalt) ground water at 250 C. Feedwater, made up to the appropriate ground-water chemistry, is pumped into an autoclave held at 250 C. Before the water passes over the specimens it flows upward through a bed of crushed basalt contained in the bottom of the autoclave. The specimens are in U-bend configuration, and each specimen contains a simulated weldment.

The oxygen fugacity is a variable in the tests. Three-month tests have been run in the past at dissolved oxygen levels of ~ 1 ppm and ~ 50 ppb. A more recent test used an oxygen level of ~ 7 ppm. Detailed weight change data for the last test are not yet available, as the specimens are currently being descaled. However, as in the past studies, the titanium alloys appear to have outstanding resistance to corrosion, and the cast irons once again appear to exhibit the highest corrosion rates.

Radiation-Corrosion Facility

To predict with any certainty what the corrosion rates of barrier materials will be under actual geologic repository conditions, it is essential that the effects of temperature, pressure, environmental chemistry, and radiation be brought to bear simultaneously on the candidate barrier material(s), which would be maintained in some appropriate flawed and stressed state.

To date, an autoclave test apparatus in a radiation facility has not been available for the program. To remedy this situation, such a facility is currently being constructed. The radiation will be provided by ^{60}Co sources in an existing Hanford Engineering Development Laboratory radiation facility. Provision is being made for isothermal heating by a steam source; dynamic flow conditions; complete chemical characterization of influent and effluent liquids; and electrochemical monitoring of the specimens and the environment within the reaction vessels.

The three autoclaves required for the facility have been ordered, and the preliminary safety assessment has been completed. The facility should be usable by July, 1981.

Slow-Strain-Rate Apparatus

A slow-strain-rate test facility is also under construction. This facility will eventually have three pressure vessels and three constant-elongation-rate loading mechanisms. Rupture of metallic specimens under slow-strain-rate conditions in repository-relevant environments, with subsequent fractography and stress-strain analysis, is expected to further define susceptibility of candidate barrier materials to environmentally induced degradation.

REFERENCE

Westerman, R. E., 1980. *Investigation of Metallic, Ceramic and Polymeric Materials for Engineered Barrier Applications in Nuclear Waste Packages*, PNL-3484, Pacific Northwest Laboratory, Richland, WA, October.

WBS 1.2.3

Project: Stabilizer Materials Program

Principal Investigator: Hanford Engineering Development Laboratory (V. J. Ferrell, R. L. Fish, and N. Wynhoff)

ONWI Project Manager: D. P. Moak

Objective

The objective is to identify, test, select, and qualify stabilizer materials for use in designing spent fuel waste packages. Emphasis is placed on selecting stabilizer materials with minimal interactions.

Progress Reported Previously

In support of the stabilizer material screening effort, all preliminary interaction, cost, and availability evaluations were completed. Criteria were established for the screening of stabilizer materials based on these preliminary analyses. Based on the criteria, the initial list of 67 potential stabilizer materials was reduced to 34 that were recommended to ONWI for further testing. The criteria used in screening included consideration of thermally induced stresses, criticality concerns, and material cost and availability.

Detailed planning for temperature limit, fill process, and prebreach disposal condition compatibility testing was completed. A test matrix was developed that emphasizes screening of materials rather than development of a data base for predicting long-term disposal effects. Procurement and preparation of test components for the temperature limit tests were ongoing last quarter.

Activities During the Reporting Period

The preliminary screening of candidate spent fuel stabilizer materials was completed and reported to ONWI in November, 1980. Based on criteria set for thermally induced stresses, material cost, and material availability, an initial list of potential stabilizers was reduced to 34 that are recommended for additional screening tests.

The spent fuel stabilizer temperature limit test was completed during this report period. Tests were conducted to establish a conservative temperature limit for spent LWR fuel during the stabilizer emplacement process. Melting of metals and metal alloys, fusing of glass stabilizers, or other possible emplacement processes may require elevation of the spent fuel temperature above levels desirable for maintaining fuel rod integrity (intact cladding) or reasonable inter-rod spacing for stabilizer flow. This test was conducted to establish the temperature below which these concerns would not exist. Stabilizers that require emplacement process temperatures above the limit will be eliminated from the screening program.

The temperature limit test specimen consisted of a helium-filled, Zircaloy-4[®] cladding section with welded end caps. A 416 stainless steel mandrel placed inside the cladding

specimen occupied the same volume ratio and simulated the thermal expansion characteristics typical of fuel in a PWR fuel rod. The slightly higher 416 stainless steel thermal expansion coefficient and cladding specimen end restraints provided for a conservative simulation. Zircaloy-4 cladding specimens were backfilled with helium to 650 psi, sealed, and tested for leaks. This pressure corresponds to a cladding stress in the specimen equivalent to the calculated end-of-life stress in three-cycle PWR spent fuel cladding at temperatures experienced in a hot-cell. Calculations were based on an end-of-life pressure of 525 psi in Turkey Point PWR fuel rods.

The approach used to simulate the pouring or emplacement of the stabilizer around spent fuel assembly rods was as follows: The cladding specimens were placed in a test capsule, preheated in a separate furnace, and then placed in a test furnace at the selected test temperature for 16 hours. Calculations were performed to determine anticipated cooldown time for a full-size waste canister. The results were used as a basis for having the test last 16 hours. While actual times above the melting point may be fewer than the 16 hours selected for this test, special processing techniques may require extended times above the melting point. Thus, 16 hours is considered to be reasonably conservative. After cooling to ambient temperature, the specimens were removed from the test capsule. The tests were conducted using different specimens at each of four temperatures ranging from approximately 500 to 650 C.

The diameter of each specimen was measured at three locations prior to each test run. The average diameter (D) was calculated to provide a basis for determining the $\Delta D/D$ percent strain that occurred during simulated stabilizer emplacement. The maximum diameter of each specimen was measured after the test, and the $\Delta D/D$ percent strain was calculated using the original average diameter as the reference.

The strain was plotted as a function of test temperature to facilitate the temperature limit determination as shown in Figure 1. The temperature limit was based on a 7.5 percent $\Delta D/D$ cladding strain criterion or failure, whichever occurred first. Since failure was not detected in any of the cladding specimens, the 7.5 percent strain criterion was used for establishing 590 C as the stabilizer emplacement process temperature limit. This was justified on the basis of stabilizer flow restriction. In PWR fuel assemblies, a reduction in rod-to-rod spacing in excess of 25 percent would be expected to significantly reduce flow of stabilizer material into the interstices between rods. A strain of 7.5 percent in the cladding specimens is equivalent to the expansion of fuel rods resulting in this 25 percent reduction in rod spacing. Unless continuous filling of the fuel assembly interstices can be assured, lithostatic loads may not be uniformly distributed across the spent fuel assembly, thus compromising one of the incentives for using solid stabilizer materials.

For purposes of screening, the emplacement process temperature limit criterion of 590 C provides a basis for accepting or rejecting stabilizer materials. The immediate impact of this screening criterion on the stabilizer program is the rejection of copper and all its alloys (casting temperature ≥ 954 C) from further consideration as stabilizers for intact spent fuel rod waste packages.

Contacts with the Nevada Nuclear Waste Storage Investigations (NNWSI), Basalt Waste Isolation Project (BWIP), and Westinghouse/Advanced Energy Systems Division (W/AESD) personnel were made to establish stabilizer performance functions and requirements. Preliminary discussions indicate that the primary function of the stabilizer should be to resist collapse of the canister due to geologic lithostatic and hydrostatic pressures. Agreement was voiced

FIGURE 1. EFFECT OF TEMPERATURE ON STRAIN OF ZIRCALOY-4 CLADDING SPECIMENS

Time at temperature = 16 hours.

TABLE 1. LOSS-ON-IGNITION (LOI) TESTS

Identification Number	Particulate	Weight Loss, percent		
F-24	Graphite	0.62	}	Weight Loss Over 2 Hours @ 550 C in Helium
F-23	Mullite	0.06		
F-25	Basalt	2.26		
F-26	Zircon	0.22		
F-27	Zirconia	0.08		
F-29	Silica Amorphous	0.25		
F-30	Silica Quartz	0.27		
F-6	Na Bentonite	7.6 — H ₂ O Analysis @ 300 C		
F-6 bentonite	{ dried additional	{ 0.6 percent	after 2	{ 0.6 percent
F-25 basalt				
	2½ hours @ 300 C		days in air	0.5 percent

that either a particulate or a solid stabilizer could perform this function. This functional requirement would eliminate gases used alone as stabilizers. Although other functions would be served as a natural consequence of using either a particulate or a solid, these functions are secondary in importance. Therefore, testing efforts should probably focus on (1) measuring properties that are related to maintaining the stabilizer's ability to perform its primary function, and (2) identifying materials that exhibit minimal interactions with the cladding and candidate canister materials. Some persons contacted expressed concern about the use of lead and its alloys because of toxicity and the use of graphite and bentonite because of potential airflow problems with fine particles in hot-cell filling operations. Concern was also expressed about the stability of Bondate[®] in a radiation field. These concerns will be addressed and evaluated during the final screening process.

Concrete "formed under elevated temperature and pressure" (FUETAP) has been added to the list of potential stabilizer materials as a result of discussions involving ORNL. The strength properties and moisture content of concrete emplaced using the FUETAP process pass the screening criteria established previously. This material will be included in the interaction screening tests.

A design review for the prebreach compatibility tests was held in October. Recommendations from that review incorporated into the test program included revision of the engineering test plan, completion of capsule stress and pressure overchecks, and establishing Quality Assurance requirements for retention of test program records.

Preparations were initiated for the prebreach fill process and disposal condition material interaction screening tests. Approximately 90 percent of the component machining was completed for these tests. All stabilizer materials have been received, and test unit fabrication is 50

percent complete. Loss-on-ignition (LOI) tests were conducted on each of the particulate stabilizers.

As shown in Table 1, only basalt and the sodium bentonite exceeded the LOI water-loss criterion of 1 percent established for use of particulate stabilizers. These particulates were dried for an additional 2½ hours at 300 C; subsequent water-content analyses indicated 0.6 percent in the bentonite and 0.5 percent in the basalt.

Future Activities

Stabilizer fill-process screening tests will be conducted, and prebreach disposal condition screening tests will be initiated. Stabilizer performance functions will be established.

WBS 1.2.3

Project: NBS Independent Measurement Standards Laboratory

Principal Investigators: National Bureau of Standards (H. T. Yolken and W. P. Reed)

ONWI Project Manager: D. P. Moak

Objective

The National Bureau of Standards, acting as the Independent Measurement Standards Laboratory (IMSL), will (1) promote, on a national and international scale, consistency and accuracy of data used in nuclear waste management; and (2) provide reference data, standard reference materials, research materials, and test methods as required. The IMSL may also support the DOE isolation program in areas outside DOE's Materials Characterization Organization. In addition, the IMSL will provide independent scientific peer review of selected activities of other organizations, including the Materials Characterization Center (MCC), and will, in accordance with established NBS policy, referee material property data and measurement procedures.

Progress Reported Previously

This report is the initial start-up report for the NBS program in Nuclear Waste Management, acting as the Independent Measurement Standards Laboratory for the Materials Characterization Organization. While significant effort has been devoted to the establishment of this program, work accomplished is largely limited to literature surveys, scientific contacts within the field, and performing management functions.

Activities During the Reporting Period**Waste Forms Leachability Mechanisms**

This task is designed to review existing efforts by others and to carry out NBS work in selected areas to improve the understanding of leaching. In addition, the effects of radiation, mechanical properties, microstructure, microchemistry, transport mechanisms, and other disruptive processes will be investigated, and support will be provided to NBS effort on leachability and waste forms.

Radiation Effects. For the first quarter of FY 81, efforts were devoted to transferring a proton Monte Carlo code from Oak Ridge National Laboratory. It is now being modified to simulate alpha particle effects. Meantime, a request was sent to Energy Systems Group, Rockwell International, for the preparation of 20 aluminate crystal samples (doped with boron). Neutron activation of these samples along with 20 boron-doped glass samples will be conducted to promote alpha-induced damage.

Phase Stability. Central to waste form research and the study of leachability mechanisms is the structural integrity of the waste forms. Experimental studies on the structural stability of

crystalline materials containing large cations, such as cesium in hollandite, have been initiated. Preliminary plans are under way to examine the composition stability limits of the hollandite structure in the systems $\text{BaO-Cs}_2\text{O-Al}_2\text{O}_3\text{-TiO}_2$, $\text{BaO-Cs}_2\text{O-MgO-TiO}_2$, and possibly $\text{BaO-Cs}_2\text{O-MgO-Al}_2\text{O}_3\text{-TiO}_2$.

Mass Transport. The objective of this subtask is to evaluate existing models of mass transport in solids and their applicability to nuclear waste forms.

A preliminary study on one of the sintered hollandite ceramics, $\text{Ba}_{1.33}\text{Al}_{2.67}\text{Ti}_{5.33}\text{O}_{16}$, was made during this period using electrical impedance measurements. From this impedance spectrum it appears that most of the impedance was from the anisotropic factors. This suggests that more careful experimental improvement and data analysis are needed.

Mechanical Properties. Since leachability depends on the surface area of the waste form that is exposed to the leachant, resistance of the waste form to fracture and microcracking is a potentially important property.

During this quarter, a plan to characterize the fracture resistance of waste forms has been formulated. Requests for specimens have been sent to a number of these laboratories, and specimens will, hopefully, be received for testing by the end of the next quarter.

Waste Form Reference Measurement Methods, Reference Materials, and Data

Development of Standardized Leach Tests, in Cooperation with the MCC. In the first quarter of FY 81, the current status of leach testing has been reviewed, with particular interest in how various parameters affect the test results. These parameters, such as temperature, pH, sample preparation, CO_2 , Eh, etc., have been prioritized for their relative effect on the leaching process. This priority list is being used in the experimental design.

Research Materials for Leach Tests. There is a need for homogeneous simulated waste materials that can be used as research materials, and ultimately as standard reference materials, to assure that all laboratories are performing the standard tests properly and that interlaboratory comparison of data is valid.

To aid in the standardization process, a highly homogeneous borosilicate glass was provided as a test material for use in the MCC-1 round robin test.

Development work has been started on the composition of the first leachability research material with a composition similar to PNL 76-68. Particularly of interest is the investigation into possible metastable phase separation, since such a process caused by thermal treatment (anneal) could have a large effect on the durability of the resultant glass.

The omission of the chromophoric elements, iron, nickel, and chromium in this formulation, results in a transparent glass, compared to the essentially black and opaque 76-68, and greatly simplifies homogeneity determinations. Samples of this composition are currently being tested to determine leach characteristics.

Phase Equilibria Data for Nuclear Waste Form Materials. To aid in the development and understanding of waste forms, particularly ceramics, an evaluated compilation of available phase equilibria information of particular interest to the nuclear waste field is being assembled.

Preliminary surveys indicate that very few real-phase equilibria data are available directly relating to the nuclear waste field.

Current volumes of *Phase Diagrams for Ceramists* are being evaluated for background information.

Container, Overpacks, Sleeves, and Backfill

Container Corrosion. Important aspects of this task include an understanding of degradation mechanisms of the canister, overpack, and sleeve materials to provide accelerated testing and long-term prediction of behavior. Plans have been formulated for research aimed at improving the understanding of localized corrosion processes. This understanding is to be used to develop test methods to assess the susceptibility to localized corrosion (pitting, stress cracking, and crevice attack) of container alloys under repository conditions. In addition, a beginning has been made in gathering and reviewing data on the corrosion of canister alloys in underground environments.

Backfill. The direction of work on backfill includes evaluation of the state of the art of existing data; capabilities and knowledge in selected areas as needed to improve the understanding of the functional mechanisms such as permeability; chemical, physical, and mechanical stabilities; swelling potential on wetting; and thermal conductivity of the backfill materials when they are emplaced in a repository for nuclear waste isolation.

REFERENCES

- Levin, E. M., and H. F. McCurdie, eds., *Phase Diagrams for Ceramists*, Vol. III, 1975. The American Ceramic Society, Columbus, OH.
- Roth, R. S., L. P. Cook, and G. W. Cleek, eds., *Phase Diagrams for Ceramists*, Vol. IV, 1980. The American Ceramic Society, Columbus, OH.

WBS 1.2.4

Project: Repository Waste Package Design Studies

Principal Investigator: Westinghouse Electric Corp. (D. Newby)

ONWI Project Manager: J. A. Carr

Objective

The objective of the program is to develop the preliminary designs for practical, easy-to-handle waste packages that will retain nuclear wastes for long periods of time and then effectively retard their release so that the natural barriers provided by the geology can reduce their transport to the point where they pose no threat to man and his environment.

Progress Reported Previously

The contract was initiated in May, 1980. Efforts in preceding quarters have been directed toward obtaining a foundation in waste package design technology; preparing the program's quality assurance plan, developing an engineered waste package system specification document, initiating waste package conceptual design efforts, and establishing and participating in interface coordination efforts with other ONWI/NWTS contractors.

A systems engineering approach was used to develop a comprehensive list of waste package functional requirements based on ONWI program criteria documents and on proposed federal regulations. These functional requirements formed the basis for the generation of design ideas that are to be matured into conceptual designs.

During the initial phase of conceptual design, a feasibility study was begun to evaluate the relative merits of each concept and to determine how adequately a design concept satisfies the design criteria. This evaluation process was used to consolidate favorable design features into one or more designs to be retained for more detailed study.

Waste package design activities during this period included the issue of a final draft of the *Waste Package Design Specification*. This specification will be baselined by ONWI and will become the guide for all waste package designs.

Engineered waste package designs were evaluated for spent fuel and defense high-level waste (DHLW) in salt. Feasibility studies were concluded on over 20 design ideas in order to select the most favorable design features to be incorporated into conceptual designs. Two basic design configurations were recognized. These are the Drywell Design shown in Figure 1 and the Self-Shielded Cask Design shown in Figure 2. Some significant features of each design are given with the illustrations.

Features

- Redundant Containment Barriers
- Dual-Function Sleeve
- Crush-Resistant
- Minimizes Underground Remote Operations
- Final Seal Weld is Hands on — Result is Flexibility in Choice of Material and Wall Thickness
- Design Can Accommodate Buffers, Fillers, Getters if Needed
- Borehole Less Than 40-in. in Diameter
- Retrievable With Same Equipment as Used for Emplacement
- Overpack Material Thin Enough to Permit Single-Pass Remote Welding
- Permits Preplacement of Some Components in Non-radiation Environment
- Package-Handling Methods Already Demonstrated

FIGURE 1. DRYWELL WASTE PACKAGE DESIGN CONCEPT FOR DHLW

705498-5A

Features

- Canistered Waste Cast in Monolithic Block
- Blocks Stacked in Mined Drift
- Spacers Used for Shielding and Thermal Control

Advantages

- Self-Shielded Waste Package
- Package Self-Contained and Totally Fabricated at the Surface
- Easy to Emplace/Retrieve
- No in-Mine Drill Holes
- Maximizes Use of Mined Areas
- Minimizes Head-Room Needed for Emplacement
- No Need for Special Surface Lag-Storage Area Following Package Fabrication
- No Need for Special Lag-Storage Following Retrieval
- Shielding Material Has Potential for Replacing One Metallic Containment Barrier

FIGURE 2. SELF-SHIELDED CASK DESIGN CONCEPT

The design approach used to produce these waste package designs is based on the following principles:

- Assure functional requirements are met in a conservative manner using multiple-function components where feasible.
- Failure of one component to perform its function shall not directly cause the failure of a containment barrier to perform its primary function.

This design approach results in greater reliability through redundancy. It also aids licensability by eliminating any functional dependency between components.

Functional Performance Descriptions (FPD), written and submitted in draft form, describe designs that implement each of these design configurations for the DHLW and intact spent fuel assemblies waste forms.

The FPDs discuss those design features that conform with the design requirements and criteria. Candidate materials for various components are also discussed, along with the degree of uncertainty which surrounds material selection. A comprehensive list of concerns and areas of further investigation was compiled in the FPDs for each component or potential component. These concerns will be resolved during the next stage of design, and appropriate design modifications will be made.

Activities During the Reporting Period

A study to assess the effects of several spent fuel waste form alternatives on repository waste package designs was initiated. The study is one of three complementary projects intended to enable the selection of the most appropriate spent fuel form for repository disposal. The study is proceeding by considering several waste package concepts developed for intact spent fuel assemblies and the impact on their design caused by changing the waste form to each of four alternatives: (1) nozzles removed; (2) disassembled and close packed; (3) chopped and immobilized; and (4) punctured, vented, and resealed. The impact is assessed by considering the package's ability to continue satisfying specific design objectives (e.g., containment, retrievability, radiation safety) as the waste form is altered. To date, a qualitative assessment of effects has been completed and, presently, quantitative information is being prepared primarily in the cost and thermal performance areas.

Future Activities

During the next quarter, designs for spent fuel and DHLW in salt will be carried to the conceptual design stage and documented with a draft Topical Report. Efforts will include analytical examination of waste package design concepts, integration of current interface information, and trade studies to optimize design features. Additional waste forms and geologies will be given initial consideration during the quarter, leading to the development of further conceptual designs in subsequent quarters.

The fuel compaction impact evaluation will be completed, and a summary report will be issued.

WBS 1.2.5

Project: High-Level Waste Form Preparation (HLWFP)

Principal Investigator: Battelle Pacific Northwest Laboratories (D. H. Siemens)

ONWI Project Manager: J. A. Carr

Objective

The objective of this program is to characterize, prepare, and ship the NWVP-2 waste log to the EMAD Facility. At EMAD, the waste log will be sealed in a storage canister and then shipped to Hanford, Washington, for storage in the Near-Surface Test Facility (NSTF).

Progress Reported Previously

The project began in November, 1980. This is the first quarterly technical progress report.

Activities During the Reporting Period**Background**

Two logs of vitrified high-level waste were produced during the Nuclear Waste Vitrification Project (NWVP) performed by PNL in the late 1970s. The second of these logs is slated for storage in the NSTF as part of the BWIP Project being conducted by Rockwell Hanford Operations for the DOE. Before the waste log is placed in the NSTF, it will be transported to the EMAD Facility in Nevada, where it will be overpacked in a second canister. Prior to shipment of the waste log to EMAD, it is being characterized to ensure that it is safe to ship and store and also to provide prestorage base data.

Characterization

Calorimetry tests indicate the waste log contains 500 watts of decay heat.

Core samples were taken at 10, 20, and 50 inches from the bottom of the waste log. All three samples consisted of homogeneous, vitrified waste. After the samples were taken, the core holes were plugged and the plugs were seal-welded.

External dimensions of the waste log taken before and after filling with vitrified waste show that no appreciable changes in dimensions occurred during processing.

Preparation

Preparing the waste log for shipment will consist of welding a lid on, installing a bail, and removing the external hardware. This work is now in progress.

Future Activities

Cask Handling

The National Lead, Inc. truck cask has been selected for transport of the waste log to the EMAD Facility. New equipment has been designed and, once approved, will be fabricated and tested. To protect the cask and waste log during shipment, a spacer will be designed, fabricated, and inserted in the cask before the waste log is installed.

SUMMARY

1.3 SITE

The objective of the Site Task is to identify and characterize sites within the contiguous United States that are suitable for construction and operation of a mined geologic repository for the long-term isolation of radioactive waste. Site investigations involve developing siting criteria and identifying and screening media and geosystems using these criteria. Evaluation of these media is being carried out by relating them to the NWTS-33(2) criteria document. These evaluations are currently in various stages of the screening process.

Monitoring of acoustic events in the Climax Spent Fuel Test shows that acoustic activity has decreased to less than one event per week. These events are believed to be associated with the thermal expansion of the rock mass caused by heat from the waste canisters.

Studies of the effects of organics in deep ground waters show that minute amounts of organics, phenols, and organic acids ranging in concentration from 2 to 12 ppb are present in water from repository depth in Swedish granite. Partitioning experiments indicate increased Kd values with increased temperature and that the Kd values for strontium, cobalt, and cesium are higher in the presence of oxalic acid.

The WRIT program continued the study of the solution mechanisms of doped glass by comparing the results of the M-IAEA and static leach tests. Also, studies of the corrosion of ancient Roman glass show a depletion of sodium to a depth of 3 to 5 microns and calcium to a depth of 1 to 2 microns.

Measurements of Kd values from strontium, cesium, and technetium were completed on a samples of unconsolidated sediment from the Oakwood Salt Dome.

Experiments to determine the properties of ^{239}Pu (IV) polymer under environmental conditions showed that its suspensions are not stable above pH 5.

The source term for spent fuel in a basalt repository developed by WRIT is being used in support of a reference site initial analysis by Pacific Northwest Laboratories. Also, the initial phase of a geochemical modeling study of the basalt ground-water system has been completed. Results indicate that waters are near equilibrium in silica and oversaturated in clay minerals.

Studies of thermodynamic properties of Np (III)-Np (IV) are beginning. Neptunium is expected to have a behavior similar to plutonium.

Well-log digitization activities have been completed. Data on 375 wells in New York State were stored in a computer data file. The digitized logs are ready to be played back in uniform format and depth intervals. Well locations are in the process of being posted on a state base map. This information will be used to analyze the hydrogeologic properties of formations within New York State.

The Gibson Dome No. 1 (GD-1) boring in the Paradox Salt Basin reached a depth of 6,090 feet after a whipstock program successfully bypassed a stuck core barrel. Geotechnical drill-stem tests in GD-1 included five successful loading and hydraulic fracture tests. Work on a

conceptual flow model involved defining the flow system for three hydrostratigraphic units: (1) below the salt, (2) within the Paradox Formation, and (3) above the salt. In general, the Paradox Formation acts as an aquitard except in faulted regions and anticlinal areas. Detailed fault maps of the northeastern half of the Shay Graben and Lockhart Basin were compiled. In aerial photographs of Lockhart Basin, 30 breccia pipes were located and 19 were mapped on the ground. The 12-station Salt Valley microearthquake network continued to operate in addition to the 7- and 5-station long-term networks situated around Gibson Dome and Elk Ridge. Seismic activity during this quarter appeared to be extremely low.

Logging of the microstratigraphy of the halite beds in the Gibson Dome core has been completed by the U.S. Geological Survey. Geologic cross sections compiled by the USGS through Elk Ridge show that faulting in the Paradox Member is present in the Kidd well (reverse) and the Kubat well (normal). Hole-to-surface resistivity measurements were made by the USGS from three source depths in drill hole GD-1 at Gibson Dome.

The total area in the Permian Basin for area characterization studies is being reduced by applying appropriate screening specifications. A final report distributing the effects of climate on erosion in the Rolling Plains and along the Cap Escarpment was published. Inorganic chemical analyses and $\sigma^{18}\text{O}$ stable isotope determinations are complete for ground-water samples from the salt dissolution zone in Briscoe, Hall, and Childress counties. A new basement structure map and the first of a series of detailed cross sections across the Amarillo uplift have been completed.

Draft Gulf Coast Salt Dome Geologic Area Characterization Reports (ACR) were transmitted to DOE/NPO. The ACRs included final drafts of Volume I (ONWI-117): *Introduction, Background and Summary*, Volume II (ONWI-118): *East Texas Study Area*, and Volume IV (ONWI-120): *Mississippi Study Area*. Since the area field studies in Louisiana were completed in the fourth quarter of FY 80, analyses of aquifer characteristics were not complete, and a first draft of Volume III (ONWI-119): *Louisiana Study Area* was submitted, with plans to update to final draft in the second quarter of FY 81.

The draft *Evaluation of Area Studies of the U.S. Gulf Coast Salt Dome Basins* (ONWI-109) was transmitted to DOE for review along with the Area Characterization Reports. Appendices to the three study areas were in preparation and will be completed early in the second quarter of FY 81.

Data analysis continues on the hydrologic data obtained in Louisiana. Topical reports, dealing with particular aspects of the investigations in detail, and well completion reports, are being prepared. Directed generic research continued in Texas on dome growth, ground-water flow, and caprock formations. Studies of borehole closure rates in salt, salt mine water origins, and chemical analyses of salt and caprock core from Rayburn's and Vacherie Domes continued in Louisiana.

The FY 81 Work Plan for the Lake Superior Region Granitic Screening Project was revised to reflect the course of work, currently limited to Minnesota, pending consultation with Wisconsin and Michigan. Literature from the Minnesota Geological Survey was catalogued.

Work continued on revising the National Site Characterization and Selection Plan (NSCSP) and the accompanying Environmental Assessment. Publishing efforts continued on the Gulf

Coast and Paradox Regional and Area Environmental Characterization Reports. The Subregional Summary Report for the Permian Region was initiated.

Argonne National Laboratories began regional environmental characterization studies in the Minnesota portion of the Lake Superior Region. Approval for Pacific Northwest Laboratories to initiate regional environmental characterization studies in the Southern Appalachians was requested from DOE. The request for proposal for Environmental Program Managers was released.

Hydrology of nearly impermeable rocks focused on the modeling of constant head injection into single fractures and the design of field test equipment. Flow in deformable fractures was found to decrease and then increase as the fracture deformed.

Work at the University of Minnesota concentrated on developing improved boundary element methods for thermomechanical analysis, and a topical report on this subject was completed. Also, sensitivity analyses of repository design parameters (extraction ratio, thermal loading, repository depth, and rock properties) are being initiated.

WBS 1.3.1**Project:** Site Baseline**Principal Investigator:** Envirosphere (J. Franco)**ONWI Project Managers:** N. J. Beskid and D. B. Shieler**Objective**

The objective of this task is to prepare drafts of geologic and environmental characterization and site selection documents which, when finalized, will form the framework for obtaining and documenting the base data necessary for site identification and characterization to be used in subsequent licensing proceedings.

Progress Reported Previously

Outlines for the Site Recommendation and the Banking and Site Selection Environmental Impact Statements were completed and sent to DOE. Outlines for the Summary and Recommendation Reports and the annotated outlines for Geologic and Environmental Characterization Reports were also completed.

Work began on the preparation of two topical reports: (1) Region Selection Criteria and Methodology Report, and (2) Regional Repository Siting Report. In the first, information on various alternative methodologies for screening and selecting regions for further study is being compiled and documented. Analysis of this information will yield a minimum number of criteria that are necessary to select a study region from among various recommended alternatives. In the second, methodologies for selecting study regions are being identified and documented.

Activities During the Reporting Period

Work continued on revising the National Site Characterization and Selection Plan (NSCSP) and accompanying Environmental Assessment (EA). The NSCSP, as revised in accordance with DOE comments and suggestions which included omitting annotated outlines and appendices on specific characterizations, along with the EA for its adoption, will be bound together and are expected to be released by DOE during the next quarter.

The two topical reports, the Region Selection Criteria and Methodology Report and Regional Repository Siting Report, have been completed in draft form. Both will undergo review and be revised during the next quarter.

WBS 1.3.2

Project: Acoustic Studies in Climax Stock

Principal Investigator: Lawrence Berkeley Laboratory (E. Majer)

ONWI Project Manager: J. O. Duguid

Objective

The experiment seeks to develop and evaluate techniques for monitoring the integrity of nuclear waste repositories and to determine the suitability of granite or hard rock as a repository medium. Microseismic (MS) and acoustic emission (AE) techniques will be employed to study the relationship between discrete rock failure and stress release relative to thermal effects, radiation degradation, mine geometry, and canister spacing.

Progress Reported Previously

The experiment began in December 1979 with the emplacement of AE sensors (piezoelectric crystals), preamplifiers, and a 15-channel transient wave form analyzer. The sensors were developed in a 15-station, three-dimensional array centered on the canister drift which would eventually house the spent fuel. Preliminary findings after some three months of monitoring prior to waste emplacement and heater turn-on and several weeks following are:

1. The repository rock has average P- and S-wave velocities of 5.5 and 3.2 km/sec, respectively, for a Poisson's ratio of 0.25, values typical for granite.
2. The rock on this scale is not homogeneous in velocity and attenuation properties, with seismic waves at frequencies between 1 kHz and 10 kHz. Ongoing observations of these properties may offer a means of mapping the progress of fracture zones or zones of alteration.
3. Discrete seismic events occur within the mine structure that are not associated with the canister area. These events seem to occur at a rate of one to three per week as discrete fractures with source dimensions of the order of centimeters, presumably in response to stresses induced by mining, drilling, and coring.
4. There was no evidence that strong ground motion within the repository, associated with nearby underground nuclear explosions, induces cracking or otherwise enhances the microfracturing process.
5. After waste emplacement and heater turn-on, AE/MS activity was detected near canisters 11 through 15. The main effect seemed to be in the concrete floor above the waste canisters, based on the ability to discriminate events in rock from the (AE/MS) activity in the concrete.

Activities During the Reporting Period

Objectives

The objectives were to (1) continue monitoring the AE/MS activity, velocity, and attenuation data within the Climax facility; (2) address questions relating to ongoing physical processes within the mine through observation of their effects on rock failure and wave propagation; and (3) analyze AE/MS signals relative to known stress perturbation (i.e., thermal, mechanical, and radiation).

Results

Except for one week (December 6 to 12), AE/MS activity continued at a very low rate (one event per week) throughout the first quarter of FY 81. This is probably due to the fact that the canisters have already peaked in their heat production and are now in the cooldown stage. However, the increased acoustic activity noted during 6 to 12 December could not be traced directly to maintenance activity within the canister drift, and it seemed to be related to stress relief north of canisters 10, 11, and 12 and south of the north heater drift below the mine floor.

A piezoelectric "pulser" that emits a 1 to 5 kHz pulse was installed during the quarter. This will be used to monitor the acoustic velocity and attenuation in the rock over the duration of the experiment. The pulser was installed in hole AE19, that is, at the east end of the south heater drift 19 feet beneath the floor of the drift. The pulser will travel paths through the south pillar, canister drift floor, and the north pillar, that is, both heated and unheated zones. Results of the first five weeks of monitoring on a daily basis showed that the velocity of the P and S waves remained nearly constant at 5.5 to 5.9 and 3.2 to 3.6 km/sec, respectively. If a significant change were to occur in rock crack content or water content, P- and S-wave velocities and/or ratio should also change. The attenuation (ratio of P-wave amplitudes) characteristics are somewhat more erratic but indicate a general increase in amplitude. This may be due to a slight increase in coupling or in water content of the rock during the winter.

Conclusions

Except for the one week of high activity (there is still a possibility that this activity was due to extensometer maintenance), the AE/MS activity has decreased to less than one event per week. Velocity and attenuation measurements were started to determine changes in rock properties. Attenuation measurements seem much more sensitive than velocity measurements.

Future Activities

Monitoring of AE/MS activity and wave propagation characteristics will continue. Emphasis will be placed on developing techniques to relate rock properties to the AE/MS activity and wave propagation characteristics.

WBS 1.3.2

Project: Shale Structure Studies

Principal Investigator: Georgia Institute of Technology (C. E. Weaver)

ONWI Project Manager: J. O. Duguid

Objective

The objective is to establish geothermometry methods for determining the thermal history of shales. A second objective is to determine the geochemical, mineralogical, and structural changes that occur in shales when they are subjected to burial temperatures between the range of 100 to 400 C.

Activities During the Reporting Period

No progress report was submitted for the period.

WBS 1.3.2

Project: Thermodynamic Properties of Actinides

Principal Investigator: Lawrence Berkeley Laboratory (N. Edelstein)

ONWI Project Manager: N. J. Hubbard

Objective

The program is investigating the thermodynamic properties of anticipated actinide and fission product ions and the compounds they may form in geologic environments so that their chemical behavior and potential migration in a nuclear waste setting can be assessed with improved capability.

Activities During the Reporting Period**Experimental Survey of the Electrochemical Properties of Actinides**

The tests of stationary and rotating disk electrodes within the isolation box have been completed. The dropping mercury electrode assembly has been constructed and tested within the box as well. Studies of the Np(III) - Np(IV) couple on various substrates will now begin. This system is expected to behave much like the Pu(III) - Pu(IV) system, but it should be more accessible to an electrochemical study, since the neptunium couple is reversible, whereas the plutonium couple is not. Further, evidence points to the absence of polymeric species in the neptunium system, a greatly simplifying factor. The stability constants for the neptunium system are expected to be very similar to those for the plutonium system.

The studies of peak deconvolution have been completed. A technical paper detailing the results, "Peak Resolution by Semiderivative Voltammetry", has been submitted for publication in *Analytical Chemistry*. With techniques such as the ones reported in that paper, more complicated systems can be studied. The U(VI) system may require such an approach.

Construction and Testing of Laser Spectroscopy Equipment

A study of Nd^{+3} ions in aqueous solution, using photoacoustic spectroscopy, was undertaken. Initial work has concentrated on improvements in cell design, particularly designs where no compression fits were used, and where leakage of electrolyte between cell leads was minimized. A piezoelectric tube is being used as both cell and detector. This detector provides excellent sensitivity. A single-beam detection limit has been observed for Nd^{+3} of about 10^{-4} M in pH 3 (HClO_4) solution, using input powers of ~ 200 mW of Rhodamine 6G radiation. Both noise and water background contributed to the detection limit, so two approaches were investigated in order to reduce the detection limit. First, more samples were taken in a single photoacoustic spectrum in order to put the noise components in a higher frequency range, which would allow their removal by methods developed earlier. Therefore 1,024 points are taken, instead of the 128 points taken by earlier versions. The scan time has been reduced from 45 minutes (due to averaging) to about 8 minutes. The Nd^{+3} in dilute aqueous solution shows fine structure, even after removal of noise.

A second improvement is the use of a double-beam arrangement for the experiment. One cell contains the Nd^{+3} solution, and the other cell contains the blank (usually H_2O). Subtraction of the signal generated from the blank from that generated by the Nd^{+3} allows study of solutions more dilute than 10^{-5} M. This experimental arrangement can also be used to study the complexation of the Nd^{+3} by CO_3^{2-} , for example. Here one would place the Nd^{+3} and CO_3^{2-} in the analytic cell, and Nd^{+3} in the blank. The resulting spectrum is just that of Nd^{+3} complexed with CO_3^{2-} .

The study of these changes in the photoacoustic system has been completed.

Detailed Study of Actinide Ion-Hydroxide Interactions

The hydrolysis of UO_2^{2+} is being studied by potentiometry. Difficulties have been experienced in using standard reference electrodes with the NaClO_4 supporting electrolyte, due to clogging of the frits. Initial studies of the UO_2^{2+} system suggest that a 1:1, 1:2, and 2:2 model explains the complexation fairly well (i.e., UO_2OH^+ , $\text{UO}_2(\text{OH})_2$, and $(\text{UO}_2)_2(\text{OH})_2^{2+}$ are present over the pH range of 3 to 5). Studies continue with varying amounts of SO_4^{2-} to investigate mixed complexes of UO_2^{2+} , SO_4^{2-} , and OH^- .

WBS 1.3.2

Project: Waste-Rock Interactions Technology (WRIT) Program

Principal Investigator: Pacific Northwest Laboratory (R. J. Serne)

ONWI Project Manager: J. B. Moody

Objective

The objectives of the WRIT program are to:

- Identify and characterize mechanisms of waste from radionuclide release and subsequent geochemical interactions with engineered barriers and natural geologic media, e.g., leaching, solubility, and sorption
- Develop and evaluate the effectiveness of models (develop a perspective on applications and limitations of each model) that predict leaching, sorption, and release processes in the repository and surrounding media
- Obtain, through verification studies and documentation, the most acceptable test methodologies for sorption, leaching, solubility limitations, and waste package-geomedia interaction studies
- Support repository site characterization and licensing specifically through data generation, analysis, and predictive model formulation.

Activities During the Reporting Period

Mechanism Studies

Waste Solution Interactions. Essentially all (only the three-year samples remain) of the static 25 and 75 C doped glass leach tests are concluded, and actinide and elemental analyses have been performed. What follows is a comparison between ^{239}Pu releases for static (25 C) and modified IAEA (room temperature) experiments. The same doped glass compositions (PNL 76-68) and surface area to volumes (0.1 cm^{-1}) were used in each test. Modified IAEA (M-IAEA) leaching of doped glass is described by Bradley, Harvey, and Turcotte (1979). Figures 1 and 2 give normalized total releases (including plateout) for ^{239}Pu in deionized water and 0.03 M NaHCO_3 ground water. The M-IAEA curves represent cumulative releases into leachates which are replenished periodically. Static data represent release into leachates which are not changed. On the right-hand vertical axis, a scale for molar concentration of ^{239}Pu has been provided which applies for static data assuming the released material is water soluble. Plateout on the leach container represented an average of 20 percent of the released ^{239}Pu in the bicarbonate leachate and an average of 25 percent in the deionized water leachate. Static release in deionized water as seen in Figure 1 is below that of M-IAEA release, conceivably due to solubility limitations. The shaded region on the concentration scale is the PuO_2 solubility at a pH of 8 in 0.0015 M CaCl_2 at room temperature (Rai, Serne, and Moore, 1980). The solubility data for pH 8 and 9 are extrapolated using data obtained over the range of pH 3 to 7. Upper and lower bounds shown for solubility data are based on least-squares confidence limits provided by Rai. Leach measurements up to 32 days indicate a stable pH of about 8 is reached

FIGURE 1. RELEASE OF ^{239}Pu FROM DOPED GLASS IN DEIONIZED WATER. M-IAEA DATA IS FROM (BRADLEY, HARVEY, AND TURCOTTE, 1979) AND SOLUBILITY DATA IS FROM (RAI, SERNE, AND MOORE, 1980)

FIGURE 2. RELEASE OF ^{239}Pu FROM DOPED GLASS IN 0.03 M NaHCO_3 LEACHATE. SOLUBILITY DATA APPLIES TO 0.0015 M CaCl_2 (RAI, SERNE, AND MOORE, 1980)

(alpha-radiolysis may lower the pH after long times, however). The maximum observed plutonium release is in good agreement with solubility constrained release as predicted by Rai's data. Figure 2 shows ^{239}Pu release in the bicarbonate solution.

The discrepancy between M-IAEA and static experiments is much more dramatic. Again, solubility limits seem to be in effect. It should be mentioned that glass corrosion in static tests continues over the entire time period for both deionized water and bicarbonate leachates as indicated by increasing levels of matrix material (sodium, boron, and silicon) in the leachates. The maximum release of Pu in NaHCO_3 (Figure 2) for the static test lies between that of PuO_2 or $\text{Pu}(\text{OH})_4$ solubility data given at the measured leachate pH value of 9. Available plutonium solubility data are for very dilute CaCl_2 , and their use for the 0.03 M bicarbonate solution (Figure 2) is probably less reliable than in the deionized water leachate (Figure 1). This may be a factor in the more marginal agreement between solubility and maximum release in Figure 2.

Glass Corrosion and Gamma Radiation Effects. Studies of the corrosion of several ancient Roman glasses approximately 1.6×10^3 years of age that were unearthed near Jalame, Israel, are in progress. Glass fragments obtained consisted of broken sections of bottles or vessels in addition to larger masses of apparently unworked glass. These ancient Roman glasses are being analyzed to determine the extent and nature of their interaction with the underground environment. Additionally, unreacted sections from the interior of larger pieces of the Roman glass have been taken for the purpose of evaluation in the same manner in which waste glasses are evaluated. The intent of these tests and analyses is to put some perspective into short-term evaluations of waste glasses. Currently, only analyses of the excavated glass surfaces have been completed. Evaluations of the durability and comparisons to waste glasses are in progress.

The excavated glass came in three distinct colors: amber, green, and blue. All were basically of the same composition, as given in Table 1 which, except for the addition of 2 to 3 weight percent Al_2O_3 , is similar to modern commercial soda lime silicate glasses. To date, the origin of the colors of the glasses has not been determined.

Exposed surfaces of the excavated glasses can be classified into at least three categories: (1) visually free of corrosion, (2) covered with white scale, and (3) exhibiting considerable iridescence. Samples of each type were examined using X-ray photoelectron spectroscopy (XPS) and secondary ion mass spectroscopy (SIMS).

Note that, although evidence for corrosion could be found, the glasses were essentially intact after 1.6×10^3 years of weathering. Glass fragments still exhibited relatively sharp edges and corners, sometimes covered with a corrosion layer, indicating that most of the mass of the material still remained.

Exposed surfaces visually free of corrosion were determined by SIMS analysis to be depleted of sodium to a depth of 3 to 5 microns and of calcium to a depth of 1 to 2 microns. No evidence for the adsorption of foreign materials could be found.

The white scale found on some surfaces was high in calcium, carbon, and oxygen, as determined by XPS analysis, and was likely to consist of calcium carbonate and calcium hydroxide. Calcium carbonate was identified in high concentrations in the surrounding soil. The use of limestone as a crucible material in the manufacture of the glass may be responsible for the presence of some of the scale. Transport of water is another possibility. The scale adhered tightly to the glass surfaces, resisting attempts at removal. It is likely that the material offered considerable protection to the glass against further aqueous corrosion.

TABLE 1. ANCIENT GLASS COMPOSITIONS
(BY WEIGHT PERCENT)

	Amber Glass	Blue Glass	Green Glass
Al ₂ O ₃	2.63	2.38	2.55
B ₂ O ₃	0.10	0.08	0.05
GaO	0.03	0.03	0.06
CaO	9.20	8.16	8.47
CuO	0.04	0.07	0.08
Fe ₂ O ₃	0.51	0.51	0.55
K ₂ O	1.03	0.78	0.7
MgO	0.70	0.50	0.59
MnO ₂	0.06	0.40	1.39
Na ₂ O	16.4	17.3	17.2
PbO	—	0.09	0.1
SiO ₂	69.8	69.1	67.2
SrO	—	0.06	0.06
TiO ₂	0.07	0.07	0.07
ZrO ₂	0.06	0.04	0.05
ZnO	—	—	0.07
	100.6	99.6	99.2

The iridescence observed on some glass surfaces was attributed to the very thin scales on the surface that could easily be removed and examined. XPS analysis showed that the front and back surfaces were high in silica with small amounts of calcium, indicating that the scales were formed by repeated wetting and drying, causing the reaction layer on the glass surfaces to spall off (a behavior observed previously on other commercial soda lime silicate glasses).

Gamma-irradiation has an effect on 76-68 glass leaching due both to enhanced interaction of water and glass because of water radiolysis and to formation of nitric acid if air is present. The effects of water radiolysis will be separated from those of nitric acid formation, and the temperature and dose rate effects on glass-water interactions in the gamma field will be determined.

Most experiments will be conducted on degassed water to eliminate nitric acid formation. Time experiments will be conducted at one dose rate (1.75×10^6 roentgens/hr) at 50, 70, and 90 C. Dose rate effects will be evaluated from 10^4 to 10^7 roentgens/hr. Currently, only 50 C time experiments have been completed, with these observations:

- The Si-O bond is preferentially attacked by products of water radiolysis, resulting in a three-fold increase in the leach rate of silicon.
- If air is present, nitric acid is formed, further enhancing silicon leach rates due to the removal of protective reaction layers on the glass. The silicon leach rate enhancement can approach one order of magnitude, depending on the amount of nitric acid formed.

- Filtering of leachates through 18 A filters shows no significant change in solution analysis for leachates containing small concentrations of elements. In large concentrations (>30 ppm Si), filtration causes significant reductions in silicon, cesium, alkaline earths, and the rare earths, suggesting colloidal silica with many elements attached. Many more experiments are needed in this area.
- If air has been removed from the leachate, the pH in a gamma field reaches 9.5 after 14 days due to enhanced leaching of the glass (in a nonirradiation environment the pH is only 7.8).

Preliminary results indicate a significant influence of temperature on the gamma-irradiation effects on glass leaching.

Rock-Radionuclide Retardation Studies

Methodology Development and Application. At PNL the transport of tritium and technetium was studied with a column of hand-crushed Westerly granite and flow rates ranging from 270 m/yr to 2,800 m/yr. The maximum effective porosity of the column was 90 percent of the total static porosity. Flow rate had a minimum effect on tritium transport, but technetium eluted less as the flow rate increased. The separation between tritium and technetium peaks, however, was less than 3 percent for all but the highest flow rate. Judging the position of a peak was difficult because of peak dispersion (spreading). The centroid (elution of 50 percent of the spike) was found more reliable for measuring retardation factors.

Retardation studies were conducted at LASL using a circulating column packed with rock. Spiked ground water circulates through until a constant tracer concentration is observed. The rock material included tuff, Climax Stock granite, and Eleana argillite. The K_d values compare very well with previously obtained column data. Contact times (26 to 49 days) and solution-to-solid ratio (20 ml of spiked ground water per gram of rock) are similar to batch sorption experimental conditions. The differences between batch and circulating column K_d values indicate that solids probably change more in batch sorption tests by grinding than in experiments where the solids are stationary, so batch shaking speeds should be one oscillation per second or less.

Retardation Mechanism Studies. At PNL, batch sorption isotherms for strontium, cesium, and technetium were completed on a sample of unconsolidated sediment from a depth of about 1,290 feet one-half mile south of the Oakwood Salt Dome, Leon County, Texas. It is fine sand (mainly quartz and feldspars) with roughly 20 to 40 percent clay-size particles (kaolinite has been identified). Synthetic ground water, prepared according to the TBEG recipe, was used to wash the sample to remove all organic matter.

Figures 3, 4, and 5 show the sorption isotherms for ^{137}Cs , ^{85}Sr , and ^{99}Tc , respectively. Nonradioactive strontium or cesium carriers were added to the ground water to maintain concentration levels. ^{99}Tc was used to maintain the concentration and $^{95\text{m}}\text{Tc}$ for radiochemical determination of the fraction adsorbed. The values for $K_d(\text{Cs})$ kept increasing with decreasing cesium concentration in solution, while $K_d(\text{Sr})$ values (Figure 4) were fairly constant below about 10^{-7} M Sr. Solution chemical analysis is not finished, but the strontium from the sample may maintain a constant solution concentration of about 10^{-7} M. Strontium K_d values are between 140 and 180 ml/g. For cesium, the K_d value depends on the solution Cs concentration

FIGURE 3. ADSORPTION ISOTHERM FOR ^{137}Cs ON OAKWOOD DOME SAMPLE, LEON COUNTY, TEXAS

FIGURE 4. ADSORPTION ISOTHERM FOR ^{85}Sr ON OAKWOOD DOME SAMPLE, LEON COUNTY, TEXAS

FIGURE 5. ADSORPTION ISOTHERM FOR ^{99}Tc ON OAKWOOD DOME SAMPLE, LEON COUNTY, TEXAS

and, because it occurs so infrequently in nature, $K_d(\text{Cs})$ will probably be greater than 100 ml/g for this sample.

Technetium (Figure 5) also shows lower K_d with a higher solution Tc concentration; however, at 10^{-7} M and 10^{-6} M, the average K_d value would be negative. The spread in data points means they are not significantly different from zero. While the data at 10^{-8} and 10^{-9} M Tc are apparently different from K_d values of zero, these are not necessarily valid. It is not yet known whether Tc retardation is anion adsorption or reduction and precipitation. A short-duration column experiment with the same sample material will find out if a K_d in the range of 1 or 2 ml/g yields a retardation factor of less than 20.

The computer code MINEQL was used at LBL to predict uranium concentrations in solutions contacting quartz and a montmorillonitic clay. Concentrations determined experimentally (as a function of pH, NaCl, and initial UO_2^{+2}) were compared with predictions based on thermodynamic data and an ion exchange mechanism included in MINEQL.

MINEQL almost always overestimates solution uranium concentrations, but seldom by more than one order of magnitude.

UO_2^{+2} is dominant at $\text{pH} \leq 5$ and is probably adsorbed by an ion exchange mechanism. Also, at higher pH values, adsorption by ion exchange may not be as important. Neutral or negatively charged species may be the dominant solution species, but adsorption is much greater than predicted by an ion exchange model. Two possible mechanisms may cause this: (1) formation of an unidentified solid phase may limit solubility, and (2) adsorption of hydrolyzed species by some mechanism other than ion exchange (or electrostatic attraction).

MINEQL has aided in making predictions under many verifiable conditions. When solubility data seem to agree and concentrations still fall below predictions, another mechanism has to explain and predict the adsorption.

Solution Chemistry

Solubilities of Key Compounds and Elements. Experiments to determine the properties of ^{239}Pu (IV) polymer under environmental conditions showed that its suspensions are not stable at pH values above 5, and hence the polymer should not be mobile in the lithosphere. Its solubility falls between that of $\text{Pu}(\text{OH})_4(\text{am})$ and of $\text{PuO}_2(\text{c})$ (Figures 6 and 7) and is <10 ppb in air-equilibrated dilute salt solutions at pH 7. The oxidation state of the soluble plutonium was essentially all plutonyl [$\text{Pu}(\text{V})$ and/or $\text{Pu}(\text{VI})$], as shown by thenoyltrifluoroacetone extraction procedures. In addition, one filtered solution with high plutonium concentration (pH 3.1), when analyzed by a spectrophotometric method (Rai, Serne, and Swanson, 1980), was found to contain $\text{Pu}(\text{V})$. The other solutions contained too little Pu to use the spectrophotometric method, but because of the differences in stability fields of $\text{Pu}(\text{V})$ and $\text{Pu}(\text{VI})$, it is most likely that the predominant oxidation state in all these solutions is $\text{Pu}(\text{V})$. $\text{Pu}(\text{V})$ is expected to be present predominantly as PuO_2^+ because (1) the relative tendency of plutonium ions to form complexes is $\text{Pu}(\text{IV}) > \text{Pu}(\text{III}) > \text{Pu}(\text{VI}) > \text{Pu}(\text{V})$; (2) the only anions present in significant amounts are Cl^- and NO_3^- , which do not form strong complexes with PuO_2^+ ; and (3) the species PuO_2^+ does not further hydrolyze between a pH of 0 and approximately 8.

The solubility of the polymer in nitrate solutions was approached from both undersaturation and oversaturation directions (Figure 6). The values indicate that the soluble species was at equilibrium with the solid phase. Polymer solubility in chloride solutions (Figure 7) was approximately 0.44 log units lower than that in nitrate solutions, suggesting that the higher redox potentials in the nitrate solutions may account for the difference; the precision of measuring redox potentials with a platinum electrode in nonpoised (unbuffered) air-equilibrated solutions is low ($\approx +50$ mV).

Since polymer particles may be amorphous at first and develop crystallinity upon aging, $\text{Pu}(\text{OH})_4(\text{am})$ and $\text{PuO}_2(\text{c})$ would be the extremes, depending on the polymer's degree of crystallinity.

Solution Chemical Environment and Chemical Species of Redox Sensitive Elements. The alpha radiation studies found that dissolved N_2 was converted to HNO_3 due to alpha radiolysis of suspensions of Pu compounds. Although the intermediate reactions of HNO_3 production are not yet known, H^+ , NO_3^- , and plutonyl ions are the end products. The rate of H^+ production is proportional to the H^+ activity of the solution.

The importance of precipitation and dissolution behavior of actinide compounds under environmental conditions has been recognized by the waste management community (Rai and Strickert, 1980) and calculated as a function of Eh and pH (Newton, 1980), limiting actinide compounds to oxides and hydroxides. The solubility of an actinide is the sum of the concentrations of all species in equilibrium with it, including all the hydrolyzed forms of the various possible oxidation states. These calculations require the various solubility products, hydrolysis constants, and redox potentials but, since none of the data are available, published estimates were used (Allard, Kipatsi, and Liljenzin, 1980). An example of the results is in Figure 8. The lines of constant solubility are a function of pH and Eh. Below the solid line, the solid phase is

FIGURE 6. PLUTONIUM SOLUBILITY OF Pu(IV) POLYMER SUSPENSIONS IN NITRATE SOLUTIONS

FIGURE 7. PLUTONIUM SOLUBILITY OF Pu(IV) POLYMER IN CHLORIDE SOLUTION

FIGURE 8. LINES OF CONSTANT LOG (SOLUBILITY VS. Eh AND pH) FOR PLUTONIUM HYDROUS OXIDES

the Pu(IV) hydrated oxide, while above the line the Pu(VI) hydroxide is stable. The dotted lines are the stability limits of water. The diagonal lines show a slope of about -0.24 (-4×0.059) and represent the plutonium half reaction;

where the contours are horizontal, the half-reaction is;

This result is consistent with the observation of Rai, Serne, and Moore (1980) that where Pu(V) is the principal solution species in contact with $\text{PuO}_2 \cdot n\text{H}_2\text{O}$, the solubility depends on the Eh but is essentially independent of pH. Where the contours are vertical, the plutonium reaction is

These parameters contribute significantly to the solubility of $\text{PuO}_2 \cdot n\text{H}_2\text{O}$:

1. The solubility product of $\text{PuO}_2 \cdot n\text{H}_2\text{O}$
2. The Pu(III)/Pu(IV) redox potential
3. The Pu(IV)/Pu(V) redox potential
4. The fifth hydrolysis constant for Pu^{4+} .

Careful measurement and control of Eh (and pH) are required for meaningful solubility measurement. Eh moderators or buffers (together with a potentiostat), or quinhydrone and Nile Blue (used in the past), or inorganic buffers (such as certain osmium complexes), or organic buffers (if reversible and noncomplexing) may be used. Newton (1980) lists several organic redox buffers which may be useful over a broad Eh-pH range (Table 2).

Control of redox potential in conjunction with the work on technetium solubility has been investigated. The Fe(II)/Fe(III) system for pH values greater than 7.5 was studied, and the response of a platinum electrode was approximately 180 mV/pH unit. This corresponds to the $\text{Fe}^{2+}/\text{Fe}(\text{OH})_3^\circ$ and/or $\text{Fe}(\text{OH})^+/\text{Fe}(\text{OH})_4^-$ couples. Between pH 6 and 7 the response was approximately 120 mV/pH unit, corresponding to the $\text{Fe}^{2+}/\text{Fe}(\text{OH})_2^+$ couple and inferring that the platinum electrode may be used to measure the redox potential of systems where the Fe(II)/Fe(III) redox reaction predominates.

Experiments evaluated a Tc(VII)/Tc(IV) redox couple as the controlling factor for Tc in solution. Using the Fe(II)/Fe(III) couple to establish a measurable potential, known amounts of TcO_4^- were added to N_2 -purged solutions (0.0 M KCl). Eh, pH, Tc, and Fe(II) measurements were taken in filtered (0.22 μm) aliquots of solution. Figure 9 summarizes the results. Two concentrations of technetium (4.2 μM below pH 8 and 0.011 μM above pH 8) were studied. In general, where the Eh value was above the theoretical $\text{TcO}_4^-/\text{Tc}(\text{OH})_4^-$ boundary, TcO_4^- was not lost from solution (regions N), or in some cases an initial marginal loss was not continuous (regions M). When the experimental potentials were below the theoretical boundary, over 90 percent of the TcO_4^- was lost (regions R).

TABLE 2. POSSIBLE ORGANIC REDOX INDICATORS

Buffer	E_{m7} (mv)(a)	pH Range
Induline Scarlet	-299	3.0 to 8.6
Rosinduline 2G	-281	4.8 to 11.4
Lissamine Blue BF	-253	1 to 11
N, N' -dimethyl Viologens	-446	8.4 to 13
Indigo Carmine	-125	<9
Pyocyanine	-34	1 to 12

(a) E_{m7} (mv) is the formal redox potential at pH 7.

FIGURE 9. SUMMARY OF EXPERIMENTAL DATA OBTAINED TO DEFINE Eh-pH BOUNDARY FOR TcO_4^- REDUCTION: (EQUILIBRIUM LINES ARE AT LOG TcO_4^- VALUES INITIALLY PRESENT EXPERIMENTALLY)

These experiments provide more definitive evidence that the $\text{Tc(VII)}-\text{Tc(IV)}$ redox

$$\log[\text{TcO}_4^-] = (\text{Eh} - 0.710 + 0.0788 \text{ pH}) / 0.0197$$

is appropriate for inclusion in geochemical models evaluating technetium migration.

Waste Package Interaction Mechanisms and Systems Performance

Direct Sampling Autoclave Tests. The results of new analyses from the direct sampling autoclave tests are in Table 3. The three tests used (1) a mini-canister (6.4-cm-high by 1.9-cm-diameter stainless steel 304L cylinder filled with 76-68 glass doped with ^{238}U) and deionized water; (2) a mini-canister in contact with Climax Stock granite and its equilibrated water; and (3) a mini-canister in contact with Umtanum basalt and its equilibrated water. All mini-canisters were submerged during the run, but the open end of the canister was exposed directly to the liquid rather than being surrounded with crushed rock media as in the earlier runs. The crushed rock material in two of the tests (10 to 20 percent of the material used in previous tests) was used to maintain rock-water equilibrium.

Table 3 shows the changes in solution composition as a function of time for selected elements. These comparisons are meaningful for the glass components (cesium, boron, molybdenum, and zirconium), because they are based on a solution volume-to-glass surface area exposed in the mini-canister that was essentially equal in each test. The trends observed were similar to those in previous tests, but some of the magnitudes differed. The redox

TABLE 3. COMPARISON OF SOLUTION COMPOSITIONS FOR SERIES 3 PACKAGE TESTS, 250 C/200 PSI, CONCENTRATIONS IN PPM

	Mini-canister Only				Mini-canister + Granite				Mini-canister + Basalt			
	Time in Weeks				Time in Weeks				Time in Weeks			
	0	5	6	8	0(a)	5	6	8	0(a)	5	6	8
DIW												
Si		312	348	375	10.8	295	308	343	26.8	565	580	585
Fe		0.5	0.6	0.4	0.3	1.0	0.1	0.2	0.8	0.1	0.1	0.2
Mg		0.1	0.2	0.5	1.4	ND	0.1	0.2	1.1	ND	ND	0.2
Ca		0.3	1.0	1.3	12.4	1.4	0.5	0.6	2.2	0.2	0.2	0.7
Na		325	400	485	11.5	285	325	469	13.1	230	263	344
K		1.4	1.3	4.4	13.2	20.3	22	32	3.5	31.3	30.4	28.4
Cs		17.0	21.0	25.5	0.0	15	17	24	0.0	3.7	3.8	5.5
B		135	161	209	0.9	134	160	234	1.6	117	140	200
Mo		82.5	103	130	0.0	43	60	120	0.2	58.5	72.5	110
Zn		1.2	0.6	0.6	0.0	1.5	0.6	0.6	0.0	1.1	0.9	0.6
pH	5.5	9.7	9.6	9.6	7.6	9.1	9.1	9.1	7.7	9.2	9.2	9.2

(a) Composition of rock-equilibrated water.

conditions were probably fairly low because pyrite, present in the granite, was not changed. Boron and molybdenum, originally present only in the glass, seem unaffected by the rock media as previously observed. The silicon release from the basalt package is considerably higher than from the other two because the basalt silicon probably comes from crushed, interstitial volcanic glass, which doesn't exist in the granite packages. More silicon is expected from 76-68 glass, but the surface area of glass exposed in the mini-canister is only about 2.8 cm². The high silicon concentrations in the basalt packages tend to cement the crushed basalt on cooling so that it is difficult to remove the canister. Cesium release, also inhibited by basalt, is not significantly affected by granite. This behavior is not completely understood.

The solution data again suggested that the rock systems were buffered by at least two alteration, hydrolysis reactions:

and,

At a given alkali concentration, the solution pH is fixed by these equilibria, and as such, they appear to be among the major alkali-pH controlling reactions between 150 and 300 C. One of the limiting factors is the stability of zoisite relative to wairakite and also relative to calcite in high CO₂ systems. These reactions may be useful for estimating the solution pH in the near-field environment for granite, basalt, or other alkali feldspar-rich rock.

Intermediate-Scale, PNL-Sandia Waste Package Test. Tests incorporating more of the barriers found in a typical waste package design for salt have been termed intermediate-scale, as they are roughly a factor of 10 larger than the mini-canister tests and roughly a factor of 10 smaller than a typical full-scale waste package. This size allows incorporation of all the barriers one wishes to study and enough material for detailed analyses after the conclusion of the test. When assembled, the waste package occupies a volume of 19 liters, and can still be handled. These tests are under way.

Source-Term Modeling

Waste Form Modeling. The release of constituents from borosilicate glasses during leaching is often said to "obey" the relation:

$$(\text{amount released}) = a_0 + a_1 t^{1/2} + a_2 t,$$

where a_0 , a_1 , and a_2 are constants, and t is elapsed time. These constants are then given the following interpretation:

a_0 : Loosely bound material at the surface releases instantly and then contributes nothing more.

a_1 : Constituents diffuse from the glass matrix.

a_2 : The glass matrix dissolves.

Actually, when the processes corresponding to a_1 and a_2 occur simultaneously, the release changes from the form $a_1 t^{1/2}$ initially to $a_2' + a_2 t$ at long times, where a_2' is related to a_1 and a_2 .

Considering the behavior after any instantaneous release (a_0), this seems to explain the observation that releases increase first with the square root of time, and then linearly with time.

A potential problem with the implied physical model, i.e., diffusion from the matrix, is that " $t^{1/2}$ " behavior is seen for large polyvalent ions, such as the actinides, which would be expected to exhibit extremely small diffusivities in the glass matrix. For example, results reported for the release of ^{137}Cs , ^{54}Mn , ^{154}Eu , $^{239+240}\text{Pu}$, and ^{244}Cm from radioactive borosilicate glass show that the amount released of each increases with the square root of time for about the same time (10 to 20 days at room temperature), or in some cases longer for plutonium and curium, before beginning to increase linearly with time (Bradley, 1978).

Also, the physical model presumes a homogeneous glass, whereas in fact borosilicate glasses are phase separated, comprising submicroscopic silicate- and borate-rich phases. The borate-rich phase, less durable chemically, may be leached away from the surface to leave a chemically more durable silicate phase. This is the basis for producing Vycor® glassware.

Because of these inconsistencies, a simple leaching model has been based on the dissolution of two phases, the less durable phase being controlled by the aqueous diffusion of dissolution products in pores left in the more durable phase, connecting the less durable phase with the surface. This presumes dissolution of the less durable phase is reversible, in the

sense that accumulation of dissolution products at the phase interface somehow stops dissolution as the aqueous concentration there reaches some equilibrium value. Presumably, the aqueous concentration of a major constituent in the phase (e.g., boron) is controlling.

If the release of constituents is represented as

$$d = (\text{release depth}) = \frac{(\text{amount released})}{(\text{area})(\text{concentration in the solid})}$$

then both the "homogeneous" and "two-phase" theories predict

$$d \rightarrow d^* + vt \text{ as } t \rightarrow \infty,$$

where v is the velocity at which the (visible) surface recedes due to dissolution, and d^* is a characteristic contribution (a constant) originating from the " $t^{1/2}$ " process. Both theories also predict

$$d \rightarrow at^{1/2} \text{ as } t \rightarrow 0.$$

Using these common characteristics, the behavior predicted by either theory can be "normalized" by comparing $d - vt$ to d^* , and the time to that for which $at^{1/2}$ equals d^* . Hence,

$$\phi = \frac{d-vt}{d^*} \text{ and } \theta = t \left(\frac{d\phi}{d\sqrt{t}} \right)^2$$

where ϕ is the normalized release and θ is the normalized time. Then both theories predict

$$\phi \rightarrow \sqrt{\theta} \text{ as } \theta \rightarrow 0,$$

$$\phi \rightarrow 1 \text{ as } \theta \rightarrow \infty.$$

The two theories are compared in this form in Figure 10, which shows they differ by no more than 5 percent. Therefore, the functional forms are so close that experimental data could be fit equally well by either theory by choosing two parameters: d^* and the initial slope of

d versus $\sqrt{t}$.

Existing leach data for borosilicate glass will be compared to establish a preference for one theory.

Source-Term Modeling. The source term for spent fuel in a basalt repository developed during the last quarter of FY 80 is being used in support of a Reference Site Initial Analysis for a basalt repository by the AEGIS program. The source term is a simplified analysis based on solubility constraints in an oxidizing environment inside the waste package, and it assumes congruent leaching of spent fuel.

Geochemical Modeling

Code Development. The WATEQ family of computer codes enables the user to perform ion speciation and solubility modeling of aqueous systems. WATEQ3 was rewritten to simplify addition of new reactions and to reduce computer dynamic memory requirements.

FIGURE 10. COMPARISON OF THEORIES FOR DIMENSIONLESS RELEASE (ϕ) VS. DIMENSIONLESS TIME (θ) FOR A HOMOGENEOUS AND TWO-PHASE GLASS, AND THE ASYMPTOTIC BEHAVIOR AS $\theta \rightarrow 0$

The computer code EQ3/EQ6 is being developed at LLL for WRIT applications. Data sets for the addition to the code of strontium, fluorine, and phosphorus were completed. They contain data for all the reasonably expected major aqueous and mineral species.

Model Applications. The initial phase of a geochemical modeling study of the basalt ground-water system of a portion of the Columbia Plateau analyzed 16 geographically diverse water samples from the upper confined aquifer system of the basalt. Dissolution of the basaltic glass and formation of metastable secondary minerals is clearly continuing, although the basalts are of Miocene age (6 to 13 million years old) and presumably have undergone a long history of alteration. Most of the samples are similar in equilibria to amorphous silica, presumably from the continuing dissolution of volcanic glass, and the waters are variably oversaturated

with regard to secondary clay minerals. Near equilibrium with allophane, an amorphic aluminosilicate, indicates that the formation rate of secondary aluminosilicate minerals lags behind the rate of volcanic glass dissolution.

In certain respects (solubility limitations of dissolved constituents by metastable solid phases such as amorphous silica, amorphous aluminosilicate, and amorphous iron hydroxide), these basalt aquifer samples are analogous to snow melt and ephemeral springs in terms of solubility-limiting phases. This is remarkable considering ground-water age dates are in the range of 10,000 to 30,000 years, and it reflects the high chemical reactivity of the basalt. This high chemical reactivity of the bedrock would influence the transport of radionuclides released from a waste repository by controlling (1) the concentration of major solute species in the ground water, and (2) the occurrence of secondary minerals which are formed and then serve as sorption substrates.

Repository and Breach Condition Support. This subtask aims to provide an input to AEGIS safety assessments by estimating probable geochemical conditions for a nuclear waste repository. A study was completed on modeling transport of natural radionuclides using a simulation of uraninite (UO_2) dissolution in a variety of ground waters.

The dominant natural radionuclides in the geosphere are uranium, thorium, ^{40}K , ^{87}Rb , radium, and radon. A demonstration of the capability to model the movement of these elements in a natural system will lend credibility to the developing ability to geochemically model a waste repository. Based on available chemical information on deep aquifers, the concentration in ground water of all natural radioactive elements, except radium, should be limited by an input rate from the host rock. This rate may vary with the element because of the type of minerals each is concentrated in and because of the effects of radioactive decay on dissolution, but, given ground-water ages, the individual rates can be determined. From a knowledge of the U:Ra ratio in the host rock and the input rate of uranium to the ground water, an average retardation coefficient for radium due to an exchange process can be determined between two or more boreholes.

The resultant geochemical model, for each of the natural radionuclides, would consist of accumulation terms that are linear with time and determined by measuring the concentration of an element at a number of selected sites for which ground-water flow direction and age are known. Radium's retardation term would be estimated assuming its input rate is proportional to that of uranium (or some other element deemed more representative). This model can be tied to a hydrology model to predict natural radionuclide concentrations across an aquifer's extent, with additional modification where necessary if the ground-water chemistry changes appreciably.

Given the inventory of a spent fuel nuclear waste repository and a dissolution rate of its contents, it would be possible to compare its radioactivity with that in a natural aquifer system.

Chemical controls influence the dissolution of the repository's primary component, uraninite. A computer simulation of uraninite dissolution in basalt, granite, and tuff ground waters, where overall concentration of uranium in solution was limited by the precipitation of secondary uraninite as the spent fuel dissolved, showed that solution concentrations of uranium do not approach maximum permissible levels, but under certain conditions the concentration may be at a level well above (10 to 100 times) what is found in normal ground water.

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WBS 1.3.2

Project: Helium Dating of Ground Water

Principal Investigator: Savannah River Laboratory (SRL) (I. W. Marine)

ONWI Project Manager: J. O. Duguid

Objective

Samples from several wells which are in proximity will be dated using helium accumulation methods. These samples should give similar age dates because of the close proximity of the wells. If there is substantial variation, the method will need to be reevaluated.

Activities During the Reporting Period

No progress report was submitted for the period.

WBS 1.3.2

Project: Osmotic Effects of Clay

Principal Investigator: Savannah River Laboratory (I. W. Marine)

ONWI Project Manager: J. O. Duguid

Objective

The objective is to determine the effects of natural osmotic membranes on natural hydraulic gradients in argillaceous formations.

Activities During the Reporting Period

No progress report was submitted for the period.

WBS 1.3.2

Project: Effects of Organics in Deep Ground Water

Principal Investigator: Battelle Columbus Laboratories (BCL)
(J. L. Means)

ONWI Project Manager: J. O. Duguid

Objective

The objective of this program is to fully evaluate the possibility that organic complexing agents from a variety of sources, including the ground water, the repository rock, or the waste itself, may form strong complexes with certain radionuclides and significantly increase transport rates from the repository.

Progress Reported Previously

This program was initiated in February, 1979. Major findings since then include the following: (1) ethylenediaminetetraacetic acid (EDTA) was identified in eight different ^{238}Pu -contaminated trench leachates from the Maxey Flats radioactive waste burial grounds in Kentucky; (2) the effect of humic and fulvic acids, carbonate, and bicarbonate on U_3O_8 dissolution rates has been experimentally measured; (3) a number of organic acids and phenols have been detected in ground-water samples from Hanford, Washington, and the Finnsjön and Sernö areas of Sweden; (4) mineral and water radionuclide partitioning analyses have been conducted to determine the effect of temperature (25 to 200 C) and oxalic acid content on the adsorption of cobalt, strontium, and cesium onto kaolinite. Results to date show that systems containing oxalic acid exhibited even greater metal adsorption by kaolinite than systems lacking oxalic acid. Also of interest is that the K_d for strontium and cobalt with kaolinite actually increased from 100 to 200 C at pH 5 to 6 under mildly reducing conditions.

Activities During the Reporting Period

Research in this program is being conducted in four separate tasks: (1) a laboratory analysis of actinide-humate complex characteristics, including the strength and colloid characteristics of such complexes; (2) mineral and water radionuclide partitioning experiments are being conducted to evaluate the effects of organic ligands on radionuclide migration rates as a function of temperature, pH, and oxidation-reduction potential; (3) six additional ground-water samples from Sweden are being analyzed by gas chromatographic-mass spectrometric (GC-MS) procedures for ligands, and the humic substances in these samples are being isolated, purified, and chemically characterized; (4) organo-actinide interactions will be investigated through thermodynamic equilibria calculations. Progress on the first three tasks above is presented below. Results on the fourth task will be reported next quarter.

Actinide-Humate Complex Characteristics

Progress was made on purifying the humic and fulvic acids to be used in subsequent analyses. Humus from an Ohio forest soil was extracted and purified using the procedures shown in Figure 1. The purified humic and fulvic acid fractions will be chemically characterized, and the study of the chemical and physical characteristics of humate complexes with uranium, thorium, plutonium, and americium will be initiated.

Radionuclide Partitioning Experiments

The system (5×10^{-4} M of each $\text{Sr}(\text{NO}_3)_2$, $\text{Cs}(\text{NO}_3)$, and $\text{Co}(\text{NO}_3)_2$ in 100 ml distilled water with 0.2 gram montmorillonite with or without 2×10^{-3} M oxalic acid, temperature varied from 25 to 250 C, pH buffered at 5.0 to 6.0, and under mildly reducing conditions) was analyzed, and the data are presented in Figure 2. The results show a distinct U-shaped adsorption curve for K_d versus temperature for all three metals. K_d is lowest in the intermediate temperatures of 100 to 150 C and increases with increasing temperatures. For strontium and cobalt, the increase in K_d is in fact quite dramatic, amounting to over an order of magnitude relative to K_d values at the lower temperatures. Also of interest is that, with the exception of only one data point, the K_d for the systems containing oxalic acid is greater than the K_d for the system containing no organic ligand, exactly the reverse of what is intuitively expected. These results are consistent with those in which kaolinite was the adsorption substrate. Further experiments will be conducted next quarter.

Organic Compounds in Ground Waters from Sweden : GC-MS Analysis

GC-MS analysis of six ground-water samples from the Finnsjön and Sternö areas of Sweden were conducted. Similar to previous GC-MS analyses on other samples from the same wells, only traces of chromatographable organic compounds with metal-complexing capacity were found. The two samples from Finnsjön borehole #4 were found to contain from less than 1 to 8 ppb of several isomers of di-tert-butyl phenol and traces (less than 1 ppb) of several other C_8 - C_{10} alkyl phenols. The two samples from Finnsjön borehole #5 contain less than 1 ppb of various alkyl phenols and hexadecanoic (palmitic) acid. Both of the samples from Sternö borehole #4 contain an interesting group of alkyl phenols in total concentrations ranging from approximately 4 to 12 ppb, and one of the samples contains a trace of decanoic (capric) acid and about 2 ppb of di-tert-butyl hydroxy benzoic acid.

Organic Compounds in Ground Waters from Sweden : Humate Analysis

Ground waters were concentrated and subjected to the separation and purification scheme shown in Figure 3. Dissolved organic carbon analyses were performed on the solutions at various stages in the separation so that the distribution of organic species could be followed. The results obtained to date show that from 4 to 33 percent of the organic carbon in ground waters from Finnsjön is precipitated by acid and is therefore humic acid. Only 1.8 to 2.8 percent of the remaining nonprecipitated organic carbon is retained within a 1,000 molecular weight exclusion limit dialysis membrane, the other 97.2 to 98.2 percent having molecular weights less than 1,000. This presents problems in purifying the small molecular weight organic species from the inorganic contaminants, a subject that will be pursued in greater depth.

FIGURE 1. FLOW CHART FOR HUMIC ACID AND FULVIC ACID SEPARATION AND PURIFICATION

FIGURE 2. K_d VS TEMPERATURE FOR 5×10^{-4} M SOLUTIONS OF COBALT, STRONTIUM, AND CESIUM WITH AND WITHOUT 2×10^{-3} M OXALIC ACID, pH 5 TO 6, MILDLY REDUCING CONDITIONS

Future Activities

Research will continue on the four tasks. In particular the actinide-humate complex analyses will begin, with emphasis on complexes of humic and fulvic acid with uranium and thorium. More radionuclide partitioning experiments at elevated temperatures will be conducted with different organic ligands and adsorption substrates. Attempts to isolate and chemically characterize small molecular weight natural organic compounds in ground waters from Sweden will continue, and the task on equilibria calculations will begin.

* Means DOC measured.

FIGURE 3. FLOW CHART FOR SEPARATION AND PURIFICATION OF HYDROPHILIC ORGANIC COMPOUNDS IN GROUND WATER

WBS 1.3.3

Project: Geologic Project Manager Program—Salina Basin

Principal Investigator: Stone & Webster Engineering Corporation (D. S. Pierce)

ONWI Project Manager: R. R. Nicks

Objective

The objective of the GPM program for the Salina Basin is to plan and manage a systematic investigation of the geologic and hydrologic characteristics of the bedded salt deposits and their enclosing rocks within the basin. This investigation will provide fundamental information necessary to evaluate the basin as a whole, and areas within it, for nuclear waste repository siting.

Progress Reported Previously

The program was initiated on 21 July 1977. A regional geological characterization was undertaken of those portions of the Michigan and Appalachian Basins underlying Michigan, Ohio, and New York. The regional study consisted entirely of a survey of the available technical literature. It was supplemented in New York and Ohio with an extensive compilation of the stratigraphy from an examination of geophysical logs from hundreds of oil and gas wells. A computer program was developed and applied to central New York for modeling multidensity ground-water flow. The principal result of these studies was the identification of areas in New York and Ohio that appear to comply with site-qualification criteria. A program was prepared outlining necessary further investigations.

Activities During the Reporting Period

There was no technical activity on the project during this period.

Future Activities

An investigation will be undertaken of whether and to what extent salt fractures heal naturally. The study will involve a review of the technical literature and visits to representative salt mines.

WBS 1.3.3

Project: Paradox Basin Geologic Exploration

Principal Investigator: Woodward-Clyde Consultants (WCC) (F. R. Conwell)

ONWI Project Manager: N. A. Frazier

Objective

The objective is to plan and conduct geologic work required to identify, investigate, characterize, and license a repository in salt.

Progress Reported Previously

This project began on 30 August 1977 and includes drilling support to USGS, a vertical seismic profiling (VSP) program, and geologic siting investigations. The USGS drilling program has been completed; site restoration is ongoing. The VSP program is in its second phase and nearing completion. The first phase of the siting investigation has been completed.

The second phase, now under way, involves area-level siting investigations in four study areas: Elk Ridge, Gibson Dome, Salt Valley, and Lisbon Valley. Studies reported previously have included geologic mapping; seismic monitoring; aerial and ground reconnaissances and evaluations of structure, physiography, topography, Quaternary deposits, and geomorphology; hydrogeologic and geotechnical testing; and a borehole drilled in the Gibson Dome study area.

Activities During the Reporting Period**Stratigraphy**

The recovery and geologic logging of core from the Gibson Dome No. 1 (GD-1) boring continued. After completion of testing in the Paradox Formation, coring resumed on 3 October 1980 at a depth of 5,550 feet. The top of the Pinkerton Trail Formation was penetrated at 5,507 feet. It is composed of dark-colored siltstone, shale, limestone, dolomite, and anhydrite. The recovery in the Pinkerton Trail Formation averaged 89 percent. The top of the Molas Formation was penetrated at 5,714.5 feet and is a red mudstone containing blocks of limestone. Recovery averaged 36 percent for the unit. The Leadville Limestone was entered at 5,861.2 feet and is a gray limestone. Recovery averaged 98 percent. Approximately 47 feet of the Leadville Formation had been penetrated when the core barrel became stuck in the hole. Coring resumed after completion of a whipstock program and the hole had reached a depth of 6,090 feet on 2 January 1981. In preparation for the completion report for the Gibson Dome No. 1 boring, the boring logs are being typed and edited for clarity and consistency.

Geotechnical Engineering

Geotechnical drill-stem tests conducted in Gibson borehole GD-1 were analyzed. The results indicate that, while the volumetric strain measured during the unloading process increases with depth, the creep strain rate does not show a similar increase. The creep strain rate was nearly equal for the two shallower test depths (3,140 and 3,625 feet) and increased by more than an order of magnitude in the deeper (4,865 feet) test. A preliminary estimate of the in situ state of stress has also been interpolated from the five successful loading and hydraulic fracture tests and from the impression images that were recovered from two test zones.

Hydrology

Hydrogeology activities centered on developing a preliminary conceptual flow model of the ground-water flow system in the Paradox Basin. Hydrochemical and potentiometric data compiled from petroleum exploration sources and data obtained from the GD-1 borehole were analyzed using computer programs that generated numerical or graphical relationships between the major and minor ions. The chloride-bromide plot of GD-1 samples from the Elephant Canyon and Honaker Trail Formations implies that the water in these formations in the Gibson Dome study area is largely connate water, or at least a diagenetic water not affected by salt dissolution. Work on the conceptual flow model during this quarter involved defining the flow system for all three hydrostratigraphic units. Below the salt, the San Juan Mountains act as an important recharge area for the Mississippian flow system. Within the Paradox Formation, high pressures predominate in the anticlinal areas, whereas lower pressures are found in the areas of bedded salt. In general, the Paradox Formation acts as an aquitard except in faulted regions and in anticlinal areas. The formations above the salt are generally composed of many shallow flow systems. Potentiometric data obtained from the GD-1 boring through the formations above the Upper Honaker Trail Formation show that the static reservoir pressure increases with depth as expected with normal hydrostatic loading. In the lower Honaker, however, subnormal pressures are found that continue down through the Paradox Formation, sometimes varying erratically.

Quaternary History

Quaternary field studies for 1980 were completed in November. Sampling of the backhoe soil pits on gravel terraces in Spanish Valley and the Gibson Dome area were completed. Soil development in multiple loess deposits was also examined in backhoe test pits and in an existing bulldozer trench near Blanding. A reconnaissance was made of the Elk Ridge area for scoping the field plans for 1981; Quaternary studies in the Gibson Dome area are 95 percent complete. Samples collected for age-dating analyses include charcoal from ponded sediments in the Elk Ridge area; fossil snails from several localities for paleontological, paleoenvironmental, or amino acid analyses; carbonate soil samples for amino acid, thermoluminescent dating and analysis of carbonate content; and paleomagnetic samples from deposits thought to be early Pleistocene or Pliocene in age. A field reconnaissance was also made of areas where additional data could be attained on incision rates of the Green, Colorado, and San Juan rivers.

Structural Geology

During this quarter, the objectives of the structural geology activity were to complete fault mapping in the field, begin field reconnaissance of collapse structures, and begin compilation of revised fault maps. A total of 65 individual faults were mapped by the end of the field season. Fault mapping is essentially 99 percent complete. Detailed fault maps of the northeast half of the Shay Graben and Lockhart Basin were compiled. Reconnaissance of collapse features such as breccia pipes, collapsed stratigraphic blocks, and explosion breccia pipes was begun, and small-scale (1:125,000) maps were prepared. In all, 30 breccia pipes were located on aerial photographs of Lockhart Basin, and 19 were mapped on the ground. The geophysical data ordered during the last quarter were received.

Seismology

The program of microearthquake monitoring initiated in the summer of 1979 continued. The 12-station Salt Valley network, which began operation in August, 1980, continued to operate in addition to the 7- and 5-station long-term networks situated around Gibson Dome and Elk Ridge. Seismic activity during this quarter appeared to be extremely low. The majority of recorded events have been located in the vicinity of the Colorado River between its confluence with the Green River and Moab. Several events occurred in the month of September along the Colorado River near Potash, indicating that the possible source of the seismicity along the Colorado River is longer than was previously observed. Regional data were collected to analyze the current seismicity in the vicinity of Capitol Reef, which is the most active source of moderate-size earthquakes close to the Paradox Basin. A fault plane solution was produced that exhibited normal fault striking in a northwest-southeast direction, indicating that the Basin and Range stress field has intruded further east into the Colorado Plateau than was previously thought.

Tectonic History

Compilation of paleotectonic maps of the Colorado Plateau region for the Mesozoic and Cenozoic eras continued. Acquisition and compilation of paleostress data and interpretation of the current stress field also continued.

Future Activities

During the next quarter, coring operations on the Gibson Dome No. 1 boring will be completed and long-term hydrogeologic testing will begin. Borehole plug testing, hydrogeologic testing, and geophysical logging operations are planned to begin in the Kubat boring in the Elk Ridge area. The Salt Valley microearthquake network will be removed during the next quarter, while operations will continue on the long-term Elk Ridge and Gibson Dome networks. Reprocessing and interpretation of seismic reflection data previously purchased will begin. Additional geophysical data may be purchased.

Work will continue on a completion report for the Gibson Dome No. 1 boring. An outline will be completed, and work will begin on the Area Characterization Report.

WBS 1.3.3**Project:** Paradox Basin Studies**Principal Investigator:** U.S. Geological Survey, Denver, Colorado (R. J. Hite, Coordinator)**ONWI Project Manager:** N. A. Frazier**Geology**
(R. J. Hite)**Objective**

The objective is to conduct geological studies in the Paradox Basin to determine which one, if any, of the many salt-core anticlines in the basin should be investigated as a potential site for a repository.

Progress Reported Previously

The project began in 1978 under the Office of Waste Isolation (OWI-Union Carbide). Cuttings from Salt Valley drill holes DOE No. 1 and No. 2 were analyzed for bromine by X-ray fluorescence. Plots of these data show that both holes penetrated the same halite bed which is overturned and nearly vertical. A method was developed to determine the water content of halite rock. This involves dissolving the halite sample in methanol and then determining the amount of water in the methanol using a Karl Fischer titrator. A methodology for determining the chemical composition of the brine inclusions in halite rocks is being developed. The distribution of trace bromine in halite from DOE No. 3 indicates this hole penetrated the same bed found in DOE No. 1 and No. 2 and that it is also overturned at this location. The determinations of water-insoluble material in the halite rock of DOE No. 3 show that it is very low, averaging about 1.5 weight percent. X-ray diffraction (XRD) examination of these residues show that they generally consist of about 95 percent anhydrite, with the remainder made up of quartz, dolomite, calcite, and trace amounts of illite. A longitudinal section through the Salt Valley anticline shows that the caprock overlying the salt is sealed off to the northwest by a thick cover of Mancos shale. Additional geologic barriers to hydrologic flow paths from the caprock are the deep and structurally closed synclines flanking the structure. Pyrolysis fluorescence work, completed on the Salt Valley samples, shows that only very small amounts of hydrocarbons are generated by additional thermal maturation. Clay mineralogy for Salt Valley is finished and shows the clays have low-exchange capacity and no structural water.

Activities During the Reporting Period

Logging of the microstratigraphy of the halite beds in the Gibson Dome core has been completed. The principal features logged are the anhydrite bands. Similar work was performed on halite cores from the nearby Cane Creek and Shafer anticlines. In addition, insoluble residues were quantified and qualified on halite rock from the latter two structures. By establishing a regional microstratigraphy for each halite bed, particularly those being considered as potential depository layers, it may be possible to demonstrate the presence or

absence of diagenetic features, structural discontinuities, and internal dissolution. In addition, such data will be useful in the advent of mine opening construction and demonstrating core loss intervals.

All bromine determinations for halite rock at Salt Valley are finished and plotted. These data, along with insoluble residue data, megascopic core descriptions, and geophysical logs, are now being used for the final structural interpretation of DOE No. 3.

Geologic cross sections through Elk Ridge and Lisbon Valley are now complete and ready for drafting. The Elk Ridge section shows that faulting in the Paradox Member is present in the Kidd well (reverse) and the Kubat well (normal). Also, an anomalous zone of complex salt flowage is present in the Mountain Fuel and Supply Milk Ranch well.

It has been suggested by investigators in West Germany that in the system, methanol + H_2O + NaCl, chemical banding may tie up part of the H_2O , and that a Karl Fischer titration of the methanol may not account for all of the H_2O in the system. Thus, methanol extraction of halite rock may give H_2O determinations that are too low. To examine this potential problem, an experiment using the extraction apparatus and halite samples with measured amounts of H_2O added has been set up. After the halite dissolves, the H_2O content of the methanol will be determined by Karl Fischer titration to see if all of the added H_2O has been recovered.

Future Activities

Mineralogic and geochemical work on the Gibson Dome core is expected to begin next quarter.

Hydrology (M. S. Whitfield)

Objective

The project has two goals: (1) reconnaissance definition of the hydrology of the Paradox Basin in Colorado and Utah, and (2) definition of the hydrology of a site in Salt Valley, Utah.

Progress Reported Previously

The Paradox Basin hydrologic project began in November, 1977. In the previous quarter, project reports were in various stages of preparation and review.

Activities During the Reporting Period

All of the reconnaissance reports (except the San Miguel report) have been drafted.

The reconnaissance reports for the Green River-Moab area, the hydrology of the Moab-Monticello area, the hydrology of the Dolores River area, and the hydrology of the Blanding-Durango area have been drafted.

The well-testing report for wells DOE Nos. 4 through 9 is undergoing revisional changes as suggested by Chief, Nuclear Hydrology Program.

Future Activities

A complete review will be made by the Chief, Nuclear Hydrology Program, of the one remaining reconnaissance report to be submitted by the Colorado District. The remaining five reports will be reviewed. Colleague review comments by the Colorado and Utah Districts for the Green River-Moab report will be incorporated. Following colleague reviews, all reports will be submitted to DOE for approval.

Electrical and Electromagnetic Surveys (R. D. Watts)

Objective

The objectives are (1) to investigate specific sites in the Paradox Basin (Salt Valley, Gibson Dome, Elk Ridge) for possible structural, material, or hydrologic difficulties for long-term waste disposal, and (2) to evaluate the effectiveness of various geophysical methods for studying structure and hydrology in the Paradox environment and for locating geologic inhomogeneities beyond the range of conventional well-logging probes.

Progress Reported Previously

Conclusions from the electromagnetic studies at Salt Valley are: (1) Slingram can detect anomalies that appear to be related to near-surface perched water pods; (2) extremely low-frequency loop-loop soundings revealed no widespread, continuous dissolution at the top of the salt, but there could be small, isolated areas below the detection limit; (3) Schlumberger sounding (DC resistivity) is probably not a good method for investigating caprock above the salt; (4) a helicopter-borne electromagnetic (EM) survey revealed long anomalies within Salt Valley that are probably controlled by caprock lithology; (5) very low frequency (VLF, 15 to 20 kHz) magnetic field tilt and ellipticity data can be used to detect shallow faults covered by the Quaternary eolian deposits that are widespread in the Paradox Basin. This is a very fast and inexpensive aid in mapping faults and fault extensions. The method's penetration is limited to a few 10s of feet; (6) seismic studies showed that the top of the salt at the site of the DOE drill holes in Salt Valley dips toward the northeast limb of the anticline and indicated the presence of blocks of sandstone that have foundered into the salt; (7) geophysical well logs from Salt Valley were successfully interpreted, demonstrating that such logs can give a positive identification of most rock types in an evaporite sequence; (8) hole-to-surface resistivity measurements were successfully made from three source depths in drill hole GD-1, Gibson Dome.

Activities During the Reporting Period

VLF magnetic field tilt angle and ellipticity measurements were obtained across Shay Graben, across a known collapse feature (breccia pipe) in Lockhart Basin, and across a circular, suspected collapse feature on Hatch Point. At Lockhart Basin and Hatch Point, apparent electrical resistivity was also measured. The Shay Graben data appear to show a parallel north-bounding fault recently discovered and mapped by Woodward-Clyde Consultants, but this finding needs confirmation by additional data on parallel lines. The data at the Lockhart Basin breccia pipe indicate a halo of anomalously high conductivity. This observation correlates with the induration of the core of the breccia pipe, a feature that often increases electrical resistivity. The data at the suspected collapse feature on Hatch Point reveal no VLF anomalies characteristic of Paradox faults. This feature needs to be studied with deeper penetrating methods, but a preliminary interpretation would be that the Hatch Point circles are not collapse features.

Preliminary interpretation of hole-to-surface resistivity data is continuing. Theoretical models (layered earth, two-dimensional, and three-dimensional) have been developed for modeling hole-to-surface resistivity data.

Future Activities

Schlumberger vertical electrical soundings will be done in Lockhart Basin and the vicinity in order to study the hydrologic and stratigraphic conditions above the salt formation. If permission can be obtained from the BLM, the same sort of electrical work will be undertaken at Elk Ridge.

Data processing will begin for analysis of the Salt Valley airborne electromagnetic survey data, using the Branch of Electromagnetism and Geomagnetism's new VAX computer. There appears to be a great deal of structural mapping information available in the data, judging from the limited products supplied by the flight contractor. Much more thorough analysis of the data will be done.

The final interpretation of Gibson Dome hole-to-surface resistivity data and field studies to determine the effect of steel drill casing on hole-to-surface resistivity measurements are planned.

Remote Sensing
(J. D. Friedman)

Objective

The objective of the Paradox Basin remote sensing project is to acquire, process, and interpret geologically a variety of satellite and aircraft image products of the region in order to assess factors in the tectonic and geomorphologic environment that might significantly affect the quality of a nuclear waste repository site. Particular attention will be given to detecting and delineating previously unmapped structures.

Progress Reported Previously

The project in its present form began in FY 79. Lineaments of the Paradox Basin were mapped at 1:400,000 from high-quality Landsat images that were planimetrically rectified and processed via USGS digital computer for patterns and textures of the topographic surface. Subsequent statistical analyses and comparison of rosette diagrams revealed an impressive correspondence of dominant azimuthal strike frequencies of the lineaments and those of geologically significant features.

Major lineaments mapped from Landsat images, not previously mapped and generally more than 20 km in length, were plotted primarily on the basis of linearity of landforms or of distinct tonal contrast with continuity over many kilometers.

The statistically dominant strike frequency of all lineaments mapped in this region, N40-50E, coincides with second-order gravity field, third-order magnetic-field, and fourth-order fault trends.

The northeast-trending lineaments, for the most part, represent (1) northeast-striking joint sets, (2) extensional, vertical dilation fractures with little or no horizontal displacement, (3) extensional high-angle faults of generally Laramide age with predominantly vertical displacement. These structures are compatible with inferred directions of least stress in the Colorado Plateau tectonic province during Laramide time.

The numerically dominant strike frequency of major lineaments (those more than 20 km in length) is N40-60W. This trend coincides with first-order strike frequencies of geologically mapped faults and fold axes, gravity, and magnetic-field trends, and is generally parallel to (1) the axes of the major salt anticlines, (2) the inferred southwestern boundary fault zone of the Uncompahgre uplift, and (3) the northwest-striking joint sets.

The location and strike of both northeast and northwest lineament sets may reflect deep-seated control by tectonic trends and discontinuities of the Precambrian surface as delineated in the gravity and magnetic fields.

Activities During the Reporting Period

During the last quarter, the following work was undertaken:

- Synthesis of a color composite multispectral scanning (MSS) scene of Paradox Basin from spectral band ratios 4/5, 4/6, and 6/7, and a geological interpretation
- Remote-sensing, field investigations, and compilation of the map, *Landforms of the 2-Degree Moab Quadrangle* at 1:250,000
- Computer processing and compilation of color-coded, digital terrain, hypsometric maps of the Grand Junction, Moab, and Cortez 2-degree quadrangles.

Future Activities

The following activities are planned:

- Completion of the report, *Tectonic Implications of Lineaments of the Northern Paradox Basin, Utah-Colorado*
- Completion of the map, *Landforms of the 2-Degree Moab Quadrangle, Utah, USGS, 1:250,000*
- Preparation of the report, *Band-Ratioed Color Composite Landsat Multispectral Scanner Image of Paradox Basin*, about 1:800,000 (includes Gibson Dome, Salt Valley, Lisbon Valley, etc.)
- Report planned for FY 81, *Evidence for Precambrian Control of Northeast-Trending Structures in the Paradox Basin, Utah and Colorado*.

Seismic Profiling (A. H. Balch)

Objective

The objectives are (1) to evaluate state of the art of downhole seismic equipment for profile mapping of interbeds in salt; (2) to utilize seismic geophysical data to characterize and differentiate subsurface lithologies and structural configurations in areas of interest; (3) to obtain a surface seismic profile at the Elk Ridge area which will be tied into the DOE Elk Ridge test well; and (4) to map the Paradox salt in this area, especially for thinning, solution cavities, and faulting, and also the subadjacent strata, including basement in the vicinity of Comb Ridge.

Progress Reported Previously

Two vertical seismic profile field sessions were conducted in 1979. Extensive reformatting, editing, analysis, and processing have been performed on these data, with the main objective of improving the signal-to-noise ratio. The desired reflections are very weak compared with the direct arrivals, multiple reflections, and a host of other types of unwanted seismic events. The processing and interpretation have been further complicated by the radiation from secondary and tertiary compressional and shear waves, caused by tube waves that are generated by the source in the source holes.

Activities During the Reporting Period

Previously, refinements to the velocity filtering process were reported; they appeared to bring out reflections from marker horizons reliably and clearly. Using the differences in arrival times of these reflections, it is believed that the dip, depth, and strike of the marker horizons have been measured.

FIGURE 1. PROCESSED VERTICAL SEISMIC PROFILES SHOOTING FROM DOE #2 TO DOE #3, HIGH-FREQUENCY VERSION, VERTICAL COMPONENT (LEFT) AND HORIZONTAL (Y) COMPONENT (RIGHT)

FIGURE 2. PROCESSED VERTICAL SEISMIC PROFILE SHOOTING FROM DOE #2 TO DOE #3, HIGH-FREQUENCY VERSION, PROCESSED TO ENHANCE SECONDARY PULSE RADIATED FROM THE BOTTOM OF THE HOLE, VERTICAL COMPONENT (LEFT) AND HORIZONTAL (Y) COMPONENT (RIGHT)

Additional processing has enhanced the reflections from other, deeper marker beds. A total of 20 additional data plots have been made, from which the final interpretations will be made. Particular attention will be paid to deeper horizons.

These processed data plots fall into two main classes. The first uses single-source recordings only, and the second set combines the information from the three-gun array. The vertical component data have been emphasized, because they are generally the best quality. However, some "y-component" data have been processed, and two data plots use the "x-component" recordings. Two different band-pass filters were used, and events have been separated on the basis of a variety of apparent velocities.

Two sample plots are shown in Figures 1 and 2.

Future Activities

Emphasis will be on finishing the interpretation by early April. A final report will contain, among other things, all of the interpretations, conclusions regarding the feasibility of the method, and recommendations for future work.

WBS 1.3.3

Project: Permian Basin Geologic Studies

Principal Investigator: Texas Bureau of Economic Geology (TBEG) (T. C. Gustavson)

ONWI Project Manager: G. D. Brunton

Objective

The objective is to characterize the geology and hydrology of the Palo Duro and Dalhart Basins of the Texas Panhandle. Bedded Permian salts contained within these basins may contain sites suitable for a repository for high-level nuclear wastes.

Progress Reported Previously

The third year of analysis was highlighted by initiation of a regional basement structural analysis, quantification of geophysical log parameters, analyses of the water content of basin salts, continuation of the petroleum source-rock analysis, hydrogeologic mapping, characterization of cores, genetic stratigraphic facies mapping, climatic-geomorphic monitoring of the Texas Panhandle, and continued investigations of the processes, timing, and rates of salt dissolution. Each study provides a basis for detailed data interpretation and future integration of research elements.

Activities During the Reporting Period

Remote Sensing, Shallow Stratigraphy, and Surficial Analyses

A final report describing the climatic controls on erosion in the Rolling Plains and along the Caprock Escarpment of the Texas Panhandle has been published.

The draft report describing the conditions under which the Wink Sink occurred has been completed. The Wink Sink is a large collapse feature that formed in June, 1980, as a result of dissolution of the Permian bedded salts in Winkler County, Texas.

The results of fracture analysis of Randall County core indicate a coherent sequence of brittle deformations above the zone of salt dissolution:

1. Formation of vertical fractures (joints) followed by fibrous gypsum mineralization with a fabric which indicates a horizontal maximum principal extension.
2. Formation of small, inclined faults dominantly with normal senses of dip-slip displacement but also with reverse senses of dip-slip displacement. For the dominant normal faults, a horizontal principal extension is also indicated.

3. Synchronous in-filling of faults and bedding surfaces with fibrous gypsum. The fabric of these veins, which have thicknesses up to 10 cm, indicates a vertical principal extension. The last two events are believed to have resulted from salt dissolution and the accompanying collapse. This conclusion allows for the presence of vertical fractures before salt dissolution. These fractures would, of course, have been available for increased fluid flow and accelerated dissolution. Current work is addressing the geometric relationship of joint zones and dissolution-related faults and veins.

Geochemistry and Hydrology

All inorganic chemical analyses and $\delta^{18}\text{O}$ stable isotope analyses are completed for samples of ground water in the salt dissolution zone in Briscoe, Hall, and Childress Counties. Deuterium and ^{14}C determinations are pending.

The first grid for the finite-element flow model of the salt dissolution zone is complete. The grid for the initial simulation is a vertical east-west cross section through Briscoe, Hall, and Childress Counties. Baseline hydrologic parameters such as flow and no-flow boundaries, permeability distributions, and aquifer geometry have also been identified and are incorporated into the data base for the model.

Basin Analysis

The distribution of a granite wash—a potential deep-basin aquifer and hydrocarbon reservoir—was documented in a series of isolith maps for Morrowan-Atokan, Desmoinesian, Missourian-Virgilian, total Pennsylvanian, and Wolfcampian strata. The thickness and distribution of the granite wash were strongly influenced by faulting adjacent to the Amarillo Uplift.

A detailed study of the granite wash and interbedded carbonates from Mobeetie Field continued. Good core and electric-log control were available from this field, which produced oil from Pennsylvanian granite wash. The structural contour and isolith maps of the field, combined with production information, indicate that trapping is caused by a combination of structural and stratigraphic features. The petrographic study indicates that diagenetic effects also influence the reservoir quality; potential reservoirs within the area of structural closure were destroyed by the early precipitation of calcite cement.

Studies of post-San Andres salt-bearing rocks and Mississippian strata and analysis of San Andres oil and gas occurrences in the study area are under way. Isopachous maps of post-San Andres Formations were completed, and net-salt maps of post-San Andres salt-bearing units were updated and revised. Studies of the interrelationship between post-San Andres units that crop out in the study area but do not contain salt have solved a number of stratigraphic questions that previously hindered mapping in the area. Mapping of potential hydrocarbon-bearing Mississippian rocks in the study area has proceeded as planned. The preparation of a map series on San Andres hydrocarbon occurrences in the study area was initiated. In addition, a geochemical study of San Andres core samples from wells drilled for TBEG by the DOE, as well as other core samples from oil and gas wells, may demonstrate the ancient history of fluid migration in porous San Andres dolomite and enable predictions as to what fluid movements may occur in the future.

Studies of salt quality interpreted from well logs have dealt largely with mudstone-salt compositions. Statistical discrimination of massive salt, mudstone-salt, and mudstone well-log responses has defined quantitative guidelines for interpreting these rocks. In addition, there has been a statistical ranking of various types and combinations of logs in terms of their usefulness for interpreting these lithologies.

Seismicity and Tectonic Environment

A basement structure map based on approximately 15 percent new data has been completed. The main differences between this and previous maps are: (1) the absence of a bounding fault on the north side of the western end of the Amarillo Uplift; (2) the increased complexity of faulting on the south side of the Amarillo Uplift; (3) the increased complexity in Hall, Donley, and Childress Counties; and (4) a lack of demonstrable faulting associated with the Oldham "dome." Completion of the first of a series of detailed cross sections across the Amarillo Uplift has revealed faulting of strata as young as upper Permian.

A regional Bouguer gravity map of the eastern Palo Duro Basin has been acquired, and one regional cross section has been converted into a geophysical model for two-dimensional modeling. Results show that basement topography and sedimentary fill cannot account for the magnitude of the gravity high associated with the Amarillo Uplift. Qualitative analysis of that anomaly indicates a mass excess at a depth of 25 to 30 km beneath the uplift.

WBS 1.3.3

Project: Geologic Project Manager (GPM) Program—Permian Basin

Principal Investigator: Stone & Webster Engineering Corporation (J. Briedis)

ONWI Project Manager: R. B. Laughon

Objective

The objectives are to plan, manage, and perform geologic and geotechnical investigations to identify and characterize one or more potential radioactive waste repository sites in bedded salts which underlie the Permian Basin, with initial interest in the Palo Duro and Dalhart Subbasins of Texas, and to support licensing of any such site selected by the Department of Energy for development of a repository.

Progress Reported Previously

This project was initiated on 10 July 1980. Previous work involved regional characterization studies, including (1) the collection of available data and published literature on the geology, geophysics, and hydrogeology of the region and (2) the development of specifications for screening the region to identify candidate areas. It was concluded that screening could be accomplished using available data on the following geologic characteristics:

- Salt thickness
- Salt depth
- Salt dissolution
- Commercial oil and gas fields.

Activities During the Reporting Period

The stratigraphic sequence in the Texas Panhandle Region contains seven major salt-bearing units of Permian age. From youngest to oldest, they are the Salado Formation, Seven Rivers Formation, upper San Andres and lower San Andres Formations, Glorieta Formation, and upper Clear Fork and lower Clear Fork Formations. These units contain salt interbedded with mudstone, siltstone, anhydrite, and dolomite. Recent work by the Texas Bureau of Economic Geology has produced isopachous maps of the net thickness of salt within each formation. Information on the thickness and extent of individual salt beds is not yet available. However, it is evident from inspection of well logs that salt beds within some of these stratigraphic units are thick and of relatively high quality.

Screening specifications were established for minimum depth (1,000 feet), maximum depth (3,000 feet), and for minimum net salt thickness in a formation (50 feet). Oil and gas fields and active salt dissolution areas are unsuitable for repository siting and were screened from the region. The resulting area recommended for further study is shown in Figure 1. It is a

FIGURE 1. AREA RECOMMENDED FOR FURTHER STUDY.

large area, suggesting that much of the Panhandle region may be underlain by salt that is suitable for repository siting.

It was determined that screening the region by increasing the minimum net salt thickness specification to 100 feet does not significantly reduce the size of the recommended area. In order to identify smaller candidate areas (on the order of 1,000 square miles in size), it will be necessary to apply a screening specification for minimum individual salt bed thickness. A 70-foot minimum thickness will be used to accommodate a repository design for vertical emplacement of waste canisters. Existing compilations of salt stratigraphy are not adequate for uniformly applying this specification; however, the needed data can probably be developed from available geophysical logs for wells in the area.

Future Activities

The following activities are planned:

- Completion of the Regional Characterization Report and Area Characterization Plan in final draft form
- Additional screening of the recommended study area using a specification for minimum thickness of individual salt beds.

The drilling and testing of two exploratory wells located in Donley and Oldham Counties, Texas, are planned to begin during the third quarter of FY 81.

WBS 1.3.3**Project:** East Texas Salt Dome Studies**Principal Investigator:** Texas Bureau of Economic Geology (TBEG) (C. W. Kreidler)**ONWI Project Manager:** O. E. Swanson**Objective**

The objectives are: to evaluate the potential of East Texas salt domes as hosts for the management of nuclear wastes; to determine whether salt domes and their environs will remain tectonically stable during the active life of nuclear wastes; and to determine whether domes will remain hydrologically stable over the same period of time.

Progress Reported Previously

The Bureau began activities in January, 1978. Initial studies concluded that three domes in East Texas (Keechi, Palestine, and Oakwood) merited further study. Since then the Bureau has recommended that Palestine Dome no longer be considered as a candidate dome because of the extensive brining that had previously occurred there. Dome-specific studies have continued on Oakwood and Keechi, as well as generic studies on the rest of the basin.

Activities During the Reporting Period**Surface Geology**

Analysis of lineaments in East Texas has shown that shallow domes in the southern part of the basin have had a significant influence on the regional lineament pattern by increasing the dispersion in orientation of the lineaments and by increasing their density. Radial and concentric fracturing and faulting of domal overburden are probable causes, but the minimum age of this deformation cannot be determined. The Elkhart-Mt. Enterprise fault zone has had no dramatic effect on the regional lineament pattern, partly because it is roughly parallel to the regional pattern. Lineaments are concentrated into a northeast-trending peak and a northwest-trending peak, each of which is bimodal. These trends are thought to result from preferential directions of fracture induced by interference folding at depth. This folding is induced by halokinesis and is reflected in the regional gravity field.

Microseismic studies are continuing in Rusk County on the Mt. Enterprise fault system.

Investigations concerning the potential risk that, with the return of continental glaciation and the lowering of sea level, breachment of a salt dome could result from river and stream entrenchment. The four major factors that need to be understood when investigating river entrenchment due to sea-level lowering during glacial cycles are as follows:

1. Evaluate any possible long-term uplift and determine its effects on entrenchment at the study area;

2. Understand the river response to sea-level fluctuations;
3. Understand the response tributaries will have with change in base level at confluence with the trunk stream;
4. Use the "worst conditions" to evaluate the potential of salt-dome breachment.

To study these factors, the terraces of the Trinity River were investigated. Corps of Engineers and Texas Highway Department borehole data of dams and bridges helped determine the entrenchment of the Trinity River and the tributaries that occur over Oakwood and Keechi Salt Domes. Preliminary results show that it is highly unlikely Keechi or Oakwood Dome will be breached within 200,000 years.

Hydrogeology

Progress in regard to the Oakwood ground-water flow model is as follows. An upper boundary condition which includes the effects of topography and vertical leakage was added to the model. Data for the boundary condition were obtained from topographic maps and data on the relationship between topography and head in the Queen City. Vertical permeability of the Reklaw aquitard will vary according to the range of values given in the literature for low-permeability materials. Sensitivity of the model to errors in aquitard permeability and other data will be tested.

Computer-generated contour plots of the preliminary simulation were produced and perfected such that they will be suitable for publication.

The simulation program (TERZAGI) was altered to output data necessary for plotting the fluid velocity vector at each nodal point. This will be particularly useful for displaying the three-dimensional velocity components in a two-dimensional plane. Work was begun on graphics software, which will plot the vector fields on an automatic plotter.

Results of the pumping tests and lab testing of the Wilcox core revealed some important relationships between Wilcox permeability (K) and electric-log resistivity (R). It appears that sands registering a resistivity of 20 ohm-m or greater fall within a very narrow range of K and are at least one order of magnitude greater in permeability than sands registering R less than 20 ohm-m. Therefore, the effective K of the Wilcox is largely dependent on distributions and interconnections of sands of $R > 20$. Three-dimensional mapping of distributions and thicknesses of these sands in the area of the Oakwood model was completed. Composite thicknesses of these sands within each layer of the model are being used to estimate large-scale heterogeneity for the model. Plug-size samples (70 in all) of the core were extracted and analyzed for grain-size distribution by the TBEG's Sedimentation Laboratory. Results are currently being computerized for use in various techniques of estimating permeability.

The Wilcox core (LETCo TOH-2AO) has been sampled for thin sections and grain-size analyses to a depth of 1,904 feet. Sandstone samples were collected approximately every 30 feet for a total of 48 samples. Thin-section point counts have been completed to a depth of 1,264 feet. The Wilcox sandstones are feldspathic litharenites and lithic arkoses; they generally contain 55 to 65 percent quartz and subequal amounts of feldspar and rock fragments. Porosity is commonly 15 to 20 percent.

Wilcox sandstones are generally poorly consolidated, but a few zones, commonly about 3 feet thick, are completely cemented by iron-rich calcite. Siderite (FeCO_3) nodules occur in both sands and shales starting at a depth of 734 feet. The presence of iron calcite and siderite indicates moderately to strongly reducing conditions exist in the Wilcox.

A study of Wilcox shales was initiated to identify any changes in clay mineralogy and exchange cation composition with depth. Regional changes observed in Wilcox ground-water chemistry may be related to diagenetic changes within the aquifer. Shale samples were taken approximately every 50 feet from 560 to 2,349 feet. A test sample was run to determine that cation exchange procedures could be performed on these samples. The cations calcium, sodium, magnesium, and potassium were all detected and measured in the test sample.

Caprock from beneath the overhang of the Oakwood Salt Dome (approximately 6,000 feet deep) was observed at the McBee Oil Company offices in Dallas, Texas. The sample appeared to be white, coarsely crystalline calcite with black, nonoriented shaly inclusions.

In water chemistry studies around Oakwood Dome, a good correlation between HCO_3 concentrations and corrected ^{14}C ages was found. This strongly suggests that the ^{14}C ages are correct.

Salt Core Studies

The results of the strain analysis on rock salt in the Oakwood core show that the intensity of strain decreases upward. Near the base of the core at 1,339 feet, the percentage shortening shows a statistically significant decrease from 48 to 37 at 1,178 feet. Above this point, the level of strain drops sharply in the "unfoliated" recrystallized salt zone just beneath the caprock. Shortening in the recrystallized zone was measured at 11 percent, indicating the presence of a weak preferred orientation not discernible by eye.

The types of strain measured are all of the flattening type (K values between 0 and 1), characterized by oblate, or pancake-shaped, grains. No other comparative data in other domes have been noted, but according to theory and experimental modeling, flattening strains of this type are typical of diapiric crests and atypical of diapiric stems, which contain constriction (cigar-like) fabrics. In order to estimate whether the unfoliated, recrystallized rock salt has been flattened against the overlying caprock by rise of domal salt from below, the K value of strain (a measure of the oblateness of the fabric) was plotted against depth. The resulting graph indicates that the recrystallized salt, although having largely lost the "memory" of its older, strongly foliated state, nevertheless has been subject to slight flattening compatible with upward movement of diapiric salt since recrystallization. The depth of this recrystallization is currently being studied by means of the intragranular fluid inclusions within recrystallized halite.

The orientations of the principal strain directions relative to the core axis in deformed rock salt at various points have been plotted on a stereogram. The X-axis, or principal stretching direction, fluctuates systematically within the foliation plane. The maximum stretching direction is downdip near the bottom of the core at 1,339 feet; it then rotates anticlockwise upward to 1,315 feet, then clockwise to the top of the salt core. These fluctuations can be explained in a number of ways, currently being tested; but as well as showing that large differential flow movements have occurred (also indicated by the geometric analysis

of macroscopic structures), they provide a good basis for estimating the directions of current and future creep of the rock salt in the crest of this dome.

During this quarter, emphasis was placed on the upper 200 cm of rock salt and on caprock immediately above the rock salt contact. The 200 cm section of rock salt is essentially unfoliated with a granoblastic texture and is comprised of:

1. A lower 175-cm interval of medium-grained (mean grain size 5 to 10 mm), granoblastic halite (the R-2 interval), which contains two disseminated anhydrite layers. One of these layers shows good evidence of transposition indicating that: (a) a strong fabric previously existed, and (b) that recrystallization was a solid-state, in situ process that did not involve dissolution and reprecipitation. The abrupt nature of the lower contact with foliated rock salt (less than 5 cm wide) suggests that the process was promoted by the presence of intercrystalline water rather than by pressure or temperature changes.
2. Overlying this is a 25-cm zone of coarse-grained (up to 50-mm grain size), granoblastic halite (the R-3 interval), which contains no anhydrite layering. The lower contact is sharp, as is the upper contact with caprock. Brine inclusions are abundant in halite of this interval. Such inclusions have not been identified within medium-grained halite of the underlying zone but are present in localized, coarse-grained halite crystals in the upper part of the R-2 interval. It is not yet known whether the coarse-grained halite represents a more advanced stage of recrystallization or whether it is halite precipitated directly from solution.

The rock salt-caprock contact is sharp but irregular. The contact comprises three distinct zones:

1. A 2 to 4-mm transition zone containing 60 to 90 percent anhydrite as euhedral, prismatic crystals within a matrix of halite;
2. A 3 to 5-mm zone containing 98 to 100 percent anhydrite, also as prismatic crystals; and
3. Granoblastic anhydrite caprock in which grain boundary adjustment has occurred to reduce the interfacial free energy of the system.

The caprock contains vertical extension fractures which have been in-filled by halite crystals growing at right angles to the vein walls. The transition zone is continuous across the openings of these fractures, indicating that anhydrite was accreted onto the base of the caprock after fracturing.

The caprock immediately above the rock salt contact is well laminated. The laminae are not clearly recognizable in thin section but are conspicuous in thick section (1 to 5 mm thick). They appear to be defined by varying concentrations of dark material, possibly of organic origin. These layers vary from planar, horizontal features to wavy and contorted layering. In places, dark layers cut across other (presumably primary) layers.

Compressed gas inclusions are present in a small zone of rock salt isolated within the caprock section. Similar inclusions have not been identified in the rock salt section.

From geochemical analysis of 38 samples of rock salt taken at approximately 5-foot intervals, it was determined that the mean anhydrite content of the 57 meters of core lies between 1.3 and 2.0 percent at the 95 percent level of confidence. Individual values range from 0.03 percent to more than 10 percent anhydrite. The only element or mineral showing an interesting variation with depth is sylvite (KCl). The mineral displays eight distinct peaks arranged in two nearly identical sets of four peaks each. Possible explanations for this feature are that stratigraphic duplication has occurred as the result of thrusting or that the peaks represent primary evaporite geochemistry. There is no evidence, however, of stratigraphic duplication or of cycles in the anhydrite content, or in any of the other elements analyzed.

Subsurface Geology

Volume relationships and lithostratigraphy of salt withdrawal basins (rim synclines) are being analyzed to determine rates of deposition in areas affected and unaffected by salt withdrawal. If one-to-one correlation is assumed between volume of fill in salt withdrawal areas and volume of salt moved, then rates of salt movement can be calculated. Based on analysis of seven isopachous intervals from the Lower Cretaceous (Glen Rose) to the Tertiary (Wilcox), peak rates of sedimentation (0.09 mm/year, 300 feet/my) occurred during the Lower Cretaceous (Glen Rose and Paluxy). Rates declined systematically to less than 0.03 mm/year (100 feet/my) during the Tertiary. Maximum sedimentation rates in active salt withdrawal areas range from 0.09 to 0.18 mm/year (300 to 600 feet/my). Calculations of the volumes of fill in salt withdrawal areas indicate peak rates of salt movement occurred in the Lower Cretaceous ($41 \text{ km}^3/\text{my}$; 10 miles³/year) and in the Upper Cretaceous ($16 \text{ km}^3/\text{my}$; 3.8 miles³/my). Minimum rates of salt withdrawal ($4.9 \text{ km}^3/\text{my}$; 1.2 miles³/my) occurred during the early Upper Cretaceous and Tertiary. Thus, rates of deposition and salt movement are approximately an order of magnitude greater in salt withdrawal areas than in areas with minimum salt movement.

Based on regional stratigraphic cross sections and net-sand maps for the Upper Jurassic-Lower Cretaceous, a higher percentage of sand facies for the Cotton Valley Formation has been mapped in the northwestern part of the East Texas Basin. The seismic line in this area shows no appreciable thickness variations prior to Cotton Valley deposition, thus suggesting that salt movement was initiated by differential loading during Jurassic time.

Future Activities

Activities in the future will be listed under the following headings:

1. Developing Methodology and a Model of Ground-Water Flow Around Salt Domes.
2. Understanding Deep Aquifer Hydrology and Its Effect on Dome Integrity.
3. Determining the Role of Caprock in Preventing Dome Dissolution.
4. Methodology and Determination of Internal Structure of Gulf Coast Salt Domes to Predict Degree of Salt Truncation and Internal Stress-Structure Complications.

5. Predicting Areas of Preferred Salt Dome Structural Stability by Regional Analysis of Salt Mobilization.
6. Tectonic Stability of the East Texas Basin.

These new headings alter the general orientation of the Bureau's program toward some more specific salt problems being addressed by ONWI.

WBS 1.3.3

Project: Gulf Coast Salt Dome Project

Principal Investigator: Law Engineering Testing Company (D. E. Pauls)

ONWI Project Manager: O. E. Swanson

Objective

The overall objective of this Geologic Project Manager program is to identify and rank candidate sites for a waste repository. Other scientific investigations have been established by ONWI to evaluate the geologic and hydrologic framework in the study areas of northwest Texas, Louisiana, and Mississippi. The GPM will use the results of these investigations to arrive at conclusions for dome priorities.

Progress Reported Previously

The Geologic Project Manager began activities on 29 August 1977. Subsequent activities included an evaluation of the geologic and hydrologic framework in the area of three salt domes in Texas (Keechi, Oakwood, and Palestine), three in Mississippi (Cypress Creek, Lampton, and Richton), and two in Louisiana (Rayburn's and Vacherie) and provision of a uniform data base for each of the eight potential domes within these areas. Data acquisition and evaluation continued in support of Area Characterization. Deep drilling operations for the Area Characterization phase were completed in Texas and Mississippi. These operations for the Area Characterization phase were completed in Texas and Mississippi. These operations consisted of drilling 6 deep wells and 1 water well in Texas and 30 deep wells and 11 water wells in Mississippi. Pump tests to determine aquifer properties were completed in Texas and Mississippi.

A total of 222 shallow borings were made in Texas, Louisiana, and Mississippi.

Styrofoam models of the salt stock have been prepared for 7 domes, based on purchased and measured gravity data and supplemented by seismic data and well control where available. High-resolution seismic data were used in conjunction with gravity data at both Cypress Creek and Richton Domes to develop detailed analyses of the shallow beds on domal flanks not well recorded by common-depth-point seismic data.

Activities During the Reporting Period

The major activities were concerned with the final evaluation and analysis of data compiled to characterize the three study areas in East Texas, North Louisiana, and Mississippi. The evaluation included surface and subsurface geological field data and the construction of maps showing the areal distribution of formations, their stratigraphic relations, structural attitude, and thicknesses. In particular, the analysis and interpretation of all shallow and deep drilling seismic and gravity data in the North Louisiana Study Area that had been delayed were finalized.

During the process of analyzing the data bearing on the interpretation of North Louisiana faulting, it was recognized that additional well-log analysis would be required to resolve this question. Accordingly, an analysis of faulting in northern Louisiana was initiated as a basis for evaluating reported faults.

During this reporting period, numerous physical properties of the aquifers from within the study area were determined. Aquifer testing has been completed in the North Louisiana Study Area and analysis of these tests incorporated in the text of the Area Characterization Report. Technical progress for this quarter dealt primarily with analysis of data from the North Louisiana Study Area and includes the following:

1. The hydraulic conductivities for the aquifers being studied in North Louisiana were determined using data from the LETCo aquifer test.
2. Potentiometric surfaces for the sands of the Sparta Formation and Wilcox Group were incorporated with data from the literature to construct an updated potentiometric surface map for these aquifers.
3. The environmental heads were determined for the LETCo wells which were screened in aquifers containing saline water.
4. A base map of fresh water for the study area was updated to include data from the LETCo wells.
5. Vertical flow times for waters from selected depths were determined for the area around Vacherie Dome using hydrologic data from LETCo Site LVH-6.
6. The geochemical character of the ground water from within the study was analyzed to determine changes over both time and space. The data base for this analysis was from both USGS and LETCo records. The surface-water geochemistry and its changes over time and space for waters from within the study area were analyzed using both USGS and LETCo field data.

In order to develop a perspective of the volume of data compiled during the Area Characterization phase and of the depth of analytical interpretation developed for the report, an appendix for each study area in East Texas and Mississippi was planned and prepared. The appendices included basic and supplementary data regarding borings, sample analyses, ground-water surface geology, exploration geophysics, and seismicity.

Future Activities

Reconnaissance level geologic maps of 4 domes will be updated. Well Completion Reports for 11 Mississippi sites, representing 41 wells, will be completed. The Louisiana Area Characterization Report Final Draft will be completed, as well as the Louisiana Data Appendix. Planning for the Location Characterization Phase studies will be initiated.

WBS 1.3.3

Project: Louisiana Salt Dome Studies

Principal Investigator: Louisiana State University (J. Martinez, coordinator)

ONWI Project Manager: O. E. Swanson

Geomechanics Studies
(R. L. Thoms)

Objective

The general objective is to monitor and analyze any possible ongoing movement that can be associated with salt domes. Thus, this study deals with the current tectonic stability of Vacherie and Rayburn's Domes in north Louisiana.

Progress Reported Previously

Geomechanics studies have included monitoring of any current ongoing movement over Vacherie and Rayburn's Domes with a complementary system of field instrumentation and numerical modeling. These studies include: (1) tiltmeters, (2) precise releveled, (3) core-hole closure monitoring, (4) seismicity monitoring, (5) microseismic monitoring, (6) numerical modeling, and (7) the planning of laser ranging systems.

Of the above-listed seven items, major emphasis has been placed on tiltmeters, precise releveled, and core-hole closure monitoring. Some work also has been performed on microseismic and seismic monitoring at Rayburn's and Vacherie Domes, respectively.

Progress reported previously includes relogging by a four-arm caliper "tool" of the Vacherie and Rayburn's core holes. This data is on hand in analog (trace) format. Tiltmeters have been maintained over both domes; however, those over Rayburn's were planned for deactivation. Numerical three-dimensional modeling of possible movement of Rayburn's Dome was under way, with planning initiated for similar modeling of Vacherie.

Activities During the Reporting Period**Tiltmeter Data Collection and Analysis**

Tiltmeters over Rayburn's Dome have been deactivated, and all equipment, except the buried tiltmeters themselves, has been removed. The tiltmeters will be retrieved in mid-1981. Data gathering is now restricted to the Vacherie site. Analysis of data is proceeding for use in a later topical report.

Numerical Modeling

Modeling of possible three-dimensional surface effects due to upward movement of Rayburn's Dome at depth is essentially complete. Figures 1 and 2 depict contour lines for changes in surface elevation due to a nonsymmetric, parabolic, upward-directed displacement field imposed at a depth of -500 meters (1,640 feet). The imposed displacement mode amplitude at -5000 meters (16,400 feet) was 2 mm (0.08 inch), with a resulting surface maximum displacement of 1.6 mm (0.06 inch). These maps are potentially useful for interpreting data monitored with field instrumentation such as tiltmeters and precise leveling.

FIGURE 1. THREE-DIMENSIONAL SURFACE DISPLACEMENT FOR NONSYMMETRIC MOVEMENT OF RAYBURN'S MODEL

FIGURE 2. CONTOUR OF SURFACE DISPLACEMENT FOR NONSYMMETRIC MOVEMENT OF RAYBURN'S MODEL

Core-Hole Closure Survey

Data obtained from the last four-arm logging of the Vacherie and Rayburn's core hole now appear to be of limited value. However, arrangements are being made to obtain the data in a more usable format.

Future Activities

Tiltmeter data analyses will continue. Installations functioning over Vacherie Dome will be modified to permit more convenient use and servicing of equipment. The three-dimensional modeling of Vacherie Dome will be initiated. Core-hole closure data will be analyzed, and a related topical report will be prepared.

Geohydrology Studies

(M. B. Kumar)

Objective

The objectives are (1) to determine the hydrologic stability of north Louisiana salt domes, and (2) to study the hydrologic isolation of mined openings in salt domes.

Progress Reported Previously

Geohydrology studies have included the drilling of seven exploratory observation wells. A possible saline plume was detected to the east and southeast of Vacherie Dome (no such plume was observed in the vicinity of Rayburn's Dome). Numerical modeling activities have been under way to simulate domal saline plumes. An investigation of possible salt springs in the dome areas has been initiated. Several existing salt mines in the Gulf Coast were studied for leaks. The chemical and isotopic analyses of the water from the leaks at the Belle Isle, Weeks Island, and Jefferson Island salt mines strongly indicate that the mine brines are formation waters. A similar investigation has been in progress for the Avery Island salt mine. A technique was developed to conduct routine measurements of ± 0.002 weight percent water in salt. Measurements of water content of several samples of salt and sandstone inclusions were made. Preparations were begun for isotopic analyses of trace water extracted from salt.

Activities During the Reporting Period

Numerical Modeling of Domal Saline Plumes

Numerical modeling activities pertaining to domal saline plumes are continuing. The manuscript, prepared on the principles of computation and the user's manual on the model developed thus far, is being reviewed. ORNL's AT123D model was used to simulate test cases and proved useful in verifying the results of the domal saline plume model. Work continued on additional parametric simulation with the saline plume model to ascertain the relative sensitivity of various significant parameters governing solute transport in porous media, namely, hydraulic conductivity, molecular diffusivity, and porosity. A preliminary examination

indicates that hydraulic conductivity is the most significant parameter influencing the extent of the saline plume and the salinity distribution within the plume. Molecular diffusivity, particularly at high values of hydraulic conductivity, appears to be of sufficient importance. The effects of porosity seem to be relatively less significant at low values of hydraulic conductivity and molecular diffusivity. However, porosity seems to have considerable influence at higher values of these parameters, particularly for long time periods of salt dissolution.

Surface Waters Salinity (Salt Springs) Study

Under the study program on the origin of salt springs in the Vacherie and Rayburn's dome areas, several warm signatures of infrared imageries were correlated with discernible hydrological features on colored aerial photos. A two-day field trip was made to the dome areas, and hydrological features (mostly stock ponds and some puddles in topographical lows) were examined. Water samples from 14 locations at Vacherie and two at Rayburn's were collected. Preliminary measurements of chlorinity in these samples were made with an Altex Monitor II (Beckman). All Vacherie samples showed low-chlorinity values with the significant exception of one sample which registered a chlorinity value of 4,000 mg/liter. This sample was collected from a puddle in a bulldozed and logged area about 1,800 feet northeast of LSU water well V-6. Further field investigations are planned.

Mine Leaks Study

Continued progress has been made on the subsurface leaks study of the Avery Island mine. The analytical data on 22 samples of the mine brines (leaks) and five samples of ground water from the aquifer overlying the dome are being evaluated. A preliminary examination of the data and the ternary plots-of-percentage reaction values of the major cations (e.g., Na^+ , Ca^{+2} , Mg^{+2} , and K^+) in the water samples suggest that most of the Avery Island brines are of meteoric origin. The significant exceptions are three mine leaks of nonmeteoric nature; these are richer in magnesium than the other mine brines studied. On the log-log plot of chloride concentration versus calcium concentration, these three leak samples plot significantly far away from the rest of the mine brines, suggestive of an unusually high enrichment of calcium. A preliminary study of the distribution of some trace elements in these waters has been made. The values of lithium:sodium, lithium:chlorine, rubidium:sodium, and rubidium:chlorine in the three brines are considerably higher (10 to over 100 times) than those of the rest of the mine leaks and the ground waters which themselves do not have significantly different ranges of those ratios. Currently, efforts are under way to reanalyze some of the mine brines to refine the data.

Trace-Water Analysis of Salt and Isotopic Studies

Some δD -measurements for the trace water in selected salt samples have been attempted to determine the origin of this water. The δD -values for the Jefferson Island, Weeks Island, Avery Island, Belle Island, and Vacherie samples range from -52 percent to +87 percent. Three runs on core samples from Vacherie yielded δ -values of +38 percent and +87 percent, while the δ -values from other samples are all negative. The poor reproducibility on δ -values from Vacherie suggests that there are still analytical problems, and these initial measurements on such very small water samples cannot be considered meaningful at this time. However, the apparent large difference in the isotopic composition of trace water in the interior salt dome (Vacherie) and the others, if correct, may be of great importance in deducing the origin of the trace water.

An extensive series of experiments and checks on the entire procedure of the trace-water analysis is under way. The hydrogen yields of carefully weighed water samples have been measured to verify that the uranium furnace is quantitatively reducing the water during the extraction procedure. Additional background measurements have verified earlier results that the small amount of background water in the apparatus is constant. A series of standards have been run to check the hydrogen isotope mass spectrometer. Extremely poor results have been obtained suggesting that a major part of the poor reproducibility may be in the mass spectrometer. Additional standards are being obtained to help define and solve this problem. It is to be stressed that isotopic measurement of trace-water components taxes all aspects of the analytical procedure. Several more months may be required to work out all the problems.

A paper on the amount of trace water in Gulf Coast salt domes is currently being prepared for publication.

Jefferson Island Mine Flooding Study

A study of the recent flooding of the Jefferson Island salt mine is planned. A series of aerial photographs of the surface features resulting from the mine flooding were taken. A program of intensive study of the flooded mine has been outlined. The activities planned include sparker surveys of the lake and the collapse feature on the salt dome, salinity measurements in the lake and the collapse feature, and a study of subsurface salinities in the aquifers associated with the salt dome.

Future Activities

The numerical modeling of domal saline plumes will be continued with emphasis on (1) sensitivity analyses of the significant parameters involved, and (2) verifications with ORNL's AT123D model. The future salt spring study will comprise further investigations of the surface waters in the Vacherie area that had anomalously high-chlorinity values during the preliminary examinations. Detailed sampling of the surface waters in the Rayburn's area is also planned. The Avery Island mine leak study will include the continued appraisal of all available data and isotopic analyses of the brines with observed unusual chemical character. Efforts will be intensified to investigate the reproducibility of the isotopic measurements on the trace water of salt. An in-depth analysis of the Jefferson Island mine flooding will be initiated with (1) sparker profiling across the lake on the dome, and (2) a compilation of subsurface data on the salinities of the adjacent aquifers.

Regional Geology (R. E. Wilcox)

Objectives

The objectives of this study are (1) to determine the petrology, mineralogy, and geochemistry of salt and caprock from Vacherie and Rayburn's domal cores; (2) to appraise the tectonic stability of salt domes in north Louisiana; and (3) to study the sizes and shapes of Vacherie and Rayburn's Salt Domes, Webster and Bienville Parishes, Louisiana.

The primary emphasis and research efforts will focus on the first objective during FY 81.

Progress Reported Previously

Studies of regional structure and stratigraphy of the north Louisiana salt-dome basin were completed in previous years, as were several seismic studies of Vacherie Salt Dome, reports on the lithology of salt cores from Vacherie and Rayburn's Domes, and chemical analyses of salt-core samples. Mathematical approaches to the inverse-density interpretation of gravity data over salt domes were developed.

Initial detailed studies of Vacherie and Rayburn's core layering using petrographic and geochemical techniques have included dissolution studies of salt and caprock samples.

Work last quarter included the selection of 161 salt and caprock samples from Vacherie and Rayburn's cores for detailed petrological and geochemical studies. In addition, exploratory atomic absorption analyses of two different salt textures were initiated.

Activities During the Reporting Period

Geochemistry and Petrology

The development of procedures for the very precise chemical analysis of salt and caprock is nearly completed. The dissolution of the samples is obtained by dissolving 0.5 gram of powdered sample in 100 ml of 0.1 normal HCl for halite samples and in 100 ml of 0.2 normal HCl for anhydrite samples. With the exceptions of traces of oxides, the samples will go into solution after three days. Procedures for the atomic absorption analysis of magnesium, sodium, calcium, potassium, and strontium have been developed and tested. It may be possible to improve the strontium analysis by using the graphite furnace. In the past, chlorine was determined argentometrically, but the end-point was very difficult to observe. A mercuric nitrate method is reported to have a more distinct end-point, and it is being evaluated and tested.

The first draft of a report on the geochemical work during FY 80 was received.

Six segments of core from each of the Vacherie and Rayburn's salt cores were being tested in the Compressed Air Energy Storage project.

Future Activities

Work will continue on the petrology and geochemical analyses of the core samples previously selected from Vacherie and Rayburn's Salt Domes.

Quaternary Studies (C. R. Kolb)

Objective

The objectives have been (1) to determine whether the Quaternary has been lifted out of place above Rayburn's and Vacherie Domes by upward movement of the salt since deposition of these sediments between 5,000 and 500,000 years ago, (2) to provide insights concerning the

hydrologic stability of the upper portions of the domes as evidenced by dissolution and subsidence and/or collapse during Quaternary time, and (3) to estimate the effects of erosion above the domes during the next 250,000 years.

Progress Reported Previously

Incomplete studies to date have shown: (1) no evidence of tectonic uplift over Vacherie Dome during Quaternary time; (2) possible minor dissolution and subsidence above Vacherie and more significant dissolution over Rayburn's during the Quaternary; and (3) that a maximum of 120 feet of erosion can be expected at both sites during the next 250,000 years.

Borings (20 to 200 feet deep) above the domes have penetrated unusually deep Quaternary deposits. These deposits suggest dissolution and collapse or subsidence of the upper part of the domes—a minor amount over Vacherie Dome during the Quaternary, and more than 100 feet over Rayburn's Dome during this period. Complicating matters are deep "anomalous" sands beneath the known Quaternary at Vacherie and a thick calcitic boulder zone above the anhydrite caprock at Rayburn's. It is uncertain whether these strata are of Quaternary or pre-Quaternary age.

Work during the preceding quarter involved a continuation of palynologic, grain-shape, and related studies to help determine the mode of origin and the age of the "anomalous" sands and the calcitic boulder zone. Rayburn's FY 80 shallow boring program was completed. Arrangements were completed to begin the FY 80 boring program at Vacherie.

Activities During the Reporting Period

Geomorphic Studies

A geomorphic comparison was made of stream-terrace preservation and dissection along Bashaway Creek with the same parameters in valleys to the north and south away from the influence of possible domal uplift. The findings support the hypothesis that, rather than being uplifted above the dome, the terrace surface may have been slightly downwarped because of dissolution and subsidence of the underlying salt.

Other

Other activities include:

1. The acquisition of 12 radiocarbon dates from carbonaceous material pertinent to the study of Louisiana terraces. (Two interesting dates were obtained from samples in the Rayburn's boulder zone. The dates turned out to be surprisingly young, i.e., 19,265 and 21,000 years before the present.)
2. Completion of a general index map of all completed Rayburn's boreholes.
3. A synthesis of Rayburn's palynological data.
4. Processing of perhaps half of the newly acquired Rayburn's cores.

Future Activities

Personnel of the Institute for Environmental Studies at LSU will work closely with LETCo personnel involved in obtaining access to areas where this year's Quaternary boring program will take place. Laboratory analyses of cored and bagged samples from Rayburn's Dome will continue. Considerable headway has been made on carbon-oxygen analyses of Rayburn's samples. Additional data acquired recently will delay completion of this study until early in 1981. Results of grain-shape and grain-surface studies from Rayburn's and Vacherie Domes will be summarized in reports during the coming quarter. Pollen studies from Rayburn's have been completed and will be reported during the quarter. Pollen studies from Vacherie will be completed during the quarter.

WBS 1.3.3

Project: Louisiana Hydrology

Principal Investigator: U.S. Geological Survey, District Office, Baton Rouge, Louisiana
(G. N. Ryals)

ONWI Project Manager: O. E. Swanson

Objective

The objective is to compile regional geohydrologic maps for the northern Louisiana Salt Dome Basin and to conduct hydrologic field studies, data collection, and analyses near Vacherie and Rayburn's Salt Domes. A regional, multilayered ground-water flow model will be designed.

Progress Reported Previously

Seventeen regional geohydrologic maps have been completed. The maps represent an area of more than 16,000 square miles in north-central Louisiana and southern Arkansas. They are intended for use in a proposed regional multilayered ground-water flow model. All major hydrologic features that might noticeably affect flow patterns are within the mapped area. Other maps and data necessary for the model cannot be made without data input from a regional test-drilling program.

Activities During the Reporting Period**Objectives**

During the past quarter, work objectives were (1) to maintain the ongoing ground- and surface-water data-collection network, and (2) to collect geologic and hydrologic data from test-drilling activities.

Results

A total of 16 test wells at five sites for area characterization have been completed. Pump tests have been completed for all 16 test wells, and all 16 wells were sampled for inorganic, radioelement, and trace-metal determinations.

Problems

The chief residual problem is the lack of an adequate test-drilling program. Progress toward the goal of developing a regional ground-water flow model has been stymied because needed data have not been available. The test-drilling program for area characterization (December 1979 to October 1980) will not provide the data needed to develop a regional ground-water flow model.

Future Activities

Data from the area characterization test-drilling program will be analyzed. Although data are not sufficient for a detailed regional, multilayered ground-water flow model, developmental work on a skeletal ground-water flow model will begin. A report presenting regional geohydrologic maps and the results of the area characterization test-drilling program will be prepared.

WBS 1.3.3

Project: Mississippi Hydrology

Principal Investigator: U.S. Geological Survey, Water Resources Division (WRD), District Office, Jackson, Mississippi (C. A. Spiers)

ONWI Project Manager: O. E. Swanson

Objective

The objective is to describe the geohydrology of the Mississippi Salt Dome Basin and nearby selected domes with respect to the direction and rate of ground-water flow.

Progress Reported Previously

The principal geohydrologic cross section through Richton and Cypress Creek Salt Domes was completed. This section conceptualizes the ground-water flow systems near the domes.

Potentiometric maps for the major aquifer systems near the domes were also completed.

Activities During the Reporting Period

Water-level data from the major aquifers near Richton and Cypress Creek Salt Domes have been used to calculate the potentiometric heads in the aquifers around the domes. Because the salinity of ground water in these aquifers ranges from less than 1,000 mg/liter to greater than 100,000 mg/liter, density corrections for the potentiometric head in each aquifer were made by converting measured water levels to equivalent fresh-water head and environmental water head. The method used to make density corrections is described by Lusczynski (1961).

Vertical differences in head were determined for the Wilcox, Sparta, Cook Mountain, and Miocene aquifers near Richton and Cypress Creek Domes. The highest potentiometric heads occur in the overlying aquifers. Calculations of vertical flow rates through this geologic section will require determination of vertical hydraulic conductivities (K) of the confining beds. It is postulated that flow through these confining units is extremely slow.

One possible avenue for vertical flow near the domes is the caprock that occurs along the flanks of the domes. Caprock extends to an average depth of 2,000 feet at Richton Dome and about 2,500 feet at Cypress Creek Dome. Measured K values in the caprock have been used to estimate possible flow rates up the caprock to aquifers at the top of the dome.

Horizontal hydraulic gradients were determined from water-level measurements in all 41 wells drilled in the area characterization phase of the study. Water level measurements were corrected for density to an equivalent "fresh-water head" for comparison.

Flow rates for the individual aquifers were calculated principally from the hydraulic gradient, porosity, and hydraulic conductivity. Hydraulic conductivity (K) was determined from aquifer tests performed in each test well. Problems encountered in determining K

values from the aquifer tests were: (1) partial penetration of the aquifer by the test well, (2) methane gas drive causing anomalous water-level readings, (3) boundary effects caused by lenticular sand aquifers, and (4) temperature changes in the water column causing density changes and thus anomalous water levels.

The generalized direction of flow was determined from potentiometric maps based on water-level data in test wells. Water-level data for the fresh-water aquifers (Hattiesburg and Catahoula) were supplemented by data from many water-supply wells in the study area. Water-level data for the saline aquifers (Cook Mountain, Sparta, and Wilcox) came from the few regional test wells that were drilled.

Future Activities

The geothermal gradient in an area near Richton and Cypress Creek Salt Domes will be determined, and the geochemical facies and their effect on ground-water flow near the domes will be analyzed.

REFERENCE

Luszczynski, N. J., 1961. "Head and Flow of Ground Water of Variable Density," *Journal of Geophysical Research*, Vol. 66, No. 12, pp. 4247-4256.

WBS 1.3.3

Project: Lake Superior Region Granitics

Principal Investigator: Sandia National Laboratories (D. W. Powers)

ONWI Project Manager: R. R. Nicks

Objective

The objective of this program is to screen a previously identified granitic region in the Lake Superior district to identify areas which are favorable for further characterization. The screening program will use only existing data; no field investigations will be conducted during region-to-area screening. As part of this project, Sandia National Laboratories will form a Regional Advisory Group of geotechnical experts to provide consultation regarding the screening process. Sandia will interface with this group and will utilize regional expertise in conducting the screening process. The final objective is to provide a report which identifies areas (within the region) that are favorable for further characterization.

Progress Reported Previously

This project began on 9 June 1980. During the remainder of the quarter, case structures were activated, literature searches were initiated, and screening criteria were reviewed. No agencies within the region were contacted pending agreements between the Department of Energy and the states of the Lake Superior district. During the fourth quarter of FY 80, literature searches and a criteria review were conducted. GEOREF, an on-line computer bibliography, was searched using key words. This has produced an extensive bibliography for use in the project that supplements a card file with references to the geotechnical literature of the Lake Superior district before 1960. More detailed screening criteria were drafted for some of the general criteria contained in NWTs-33(2). Literature was ordered from the Minnesota Geological Survey.

Activities During the Reporting Period**Objective**

The objective during the first quarter of FY 81 was to continue literature searches and the criteria review pending DOE agreements with states in the Lake Superior district. The Work Plan for FY 81 was to be revised and submitted to reflect alternative courses of action, schedule, and costs depending on agreements with the other states in Lake Superior district.

Procedures

The literature from the Minnesota Geological Survey was catalogued, and review of the sources has begun.

The Work Package Authorization Systems (WPAS) for FY 81 was revised to reflect the course of work depending on whether agreements are reached between DOE and the states of Wisconsin and Michigan to conduct the screening program. The WPAS indicates that a relatively constant level of effort can be maintained, pending receipt of funding authority. Adding Wisconsin and Michigan to the survey will extend the time to perform the task.

Results

The WPAS was submitted for approval, and the review of documents continues. Minnesota Geological Survey documents in draft form were recommended for publication with support from ONWI through Sandia or directly.

Conclusions

The project is moving at the appropriate pace pending DOE agreements with the states and funding authorization.

Future Activities

Supplementary orders will be sent to various organizations or agencies. Draft criteria for screening will be completed, reviewed, and revised internally.

REFERENCE

Office of Nuclear Waste Isolation, 1980. *NWTS Criteria for the Geological Disposal of Nuclear Wastes: Site-Qualification Criteria*, NWTS-33(2), Battelle Memorial Institute, Columbus, OH.

WBS 1.3.3

Project: Well-Log Digitization, Central New York

Principal Investigator: Syracuse University (J. E. Robinson)

ONWI Project Manager: R. R. Nicks

Objective

The major objective of this program is to digitize, correct, and compile in a computer data file previously obtained information on the geologic structure, deep hydrologic systems, and potential for petroleum and natural gas within Central New York State so that it may be analyzed for rock porosity, fluid content, and fluid flow using techniques currently employed in the petroleum industry.

Progress Reported Previously

The program began in July, 1979, when equipment was ordered. During the first two quarters, the equipment was purchased and installed, operating personnel were hired, and processing programs were written. During the third quarter, interfacing and equipment bugs were worked out and log digitizing was started. Production log digitizing continued throughout the fourth and fifth quarters, with some 278 suites of well logs digitized.

Activities During the Reporting Period**Objectives**

The objectives were: (1) to finish digitizing suitable logs in the project area; (2) to complete the preliminary playback, check, and correction of all digitized logs; (3) to set standardization parameters for gamma and neutron logs and plot standardized logs; (4) to transcribe original standardized logs on individual tapes; and (5) to draft cross sections used in log standardization.

Procedures

Logs from 97 more wells were digitized. This makes a total of 375 digitized wells, which is almost 50 percent more than the minimum contract number.

The digitized logs have been checked and corrected. Digital logs are being converted from the original recordings, which were a function of log-curve displacement, to values computed for uniform 6-inch increments of depth.

Clean formations have been picked and standardization parameters set for over 200 of the digitized wells.

Cross sections used in standardization of log parameters have been drafted and are ready for transmission.

Digitized well locations and American Petroleum Institute (API) numbers have been indicated on the 1:250,000 base map of the area provided by Stone and Webster; however, recent wells are not included on this base map. The recent wells are being located and posted from the individual New York State well-location maps.

Computer programs for transcribing and presenting the log data are operational. Improvements are incorporated into the computer programs as they are conceived and tested.

Results

The digitizing equipment continued to function, and the digitization of well logs has been completed. Digitized logs are ready to be played back in uniform format and in uniform depth intervals.

Conclusions

Capture of the basic data is completed, with nearly 50 percent more well logs being located and digitized than originally anticipated. Transcription to the final form is under way.

Future Activities

Six activities are planned to complete this project: (1) finish the playback and convert all the data to uniform depth intervals; (2) calculate and play back the standardized logs; (3) prepare the tapes and hard copy for transmission to the DOE; (4) complete the 1:250,000 base map showing the location and API number of all digitized wells; (5) complete the revised Quality Assurance Plan according to what has been learned from all facets of the program; and (6) complete the final report.

WBS 1.3.5

Project: Environmental Characterization

Principal Investigator: NUS Corporation (J. DiNunno, W. Belter)—
Salina and Permian

Bechtel Group, Inc. (N. Norman)—Gulf Coast and Paradox

Environmental Projects Manager (none)—Granitic Regions

ONWI Project Managers: NUS—D. J. Guzzetta
BGI—R. B. McPherson
EPM—N. J. Beskid

Objective

The objective is to conduct environmental studies at the regional, area, and location levels as a means of identifying environmental aspects of concern or interest in relation to the geological formation of interest. The studies provide a means of identifying sites that may be potentially the most suitable for a repository as well as documenting the base data to be used in subsequent licensing documents such as Environmental Reports.

Progress Reported Previously

The characterization tasks were begun on 7 July 1977.

Three proposals to perform regional environmental studies in granitic regions were received from ANL, PNL, and ORNL. The three laboratories were assigned specific granitic regions and final proposals requested.

A meeting with the state of Minnesota was held on September 23 to discuss initiation of regional studies in the Minnesota portion of the Lake Superior Region.

DOE approved the final regional environmental characterization report, the final regional characterization summary and area recommendation report, and the draft area characterization reports for the Gulf Coast.

All long lead-time equipment has been procured by BGI in preparation for field environmental baseline studies. Meteorological shelters and other equipment for field work were assembled.

Briefings on the status of studies in the Paradox Basin Study Region were given to the BLM in Monticello, Utah, during July and to the public in Moab and Blanding, Utah, on 3 and 4 September. DOE approved the final regional environmental characterization report,

the draft regional characterization summary and area recommendation report, and the draft area environmental characterization report. Studies on water availability and rail access in Southeastern Utah were initiated by Bechtel.

No progress was made in the Salina and Permian Basins.

No environmental field activities were conducted in any region.

Activities During the Reporting Period

Publishing efforts continued on the Gulf Coast environmental characterization reports. The regional and regional summary reports are expected early February. The draft area environmental characterization reports are expected at the end of March. A meeting was held with Bechtel to initiate work on the Gulf Coast Site Characterization Plan. The final draft report of the transportation study that estimates doses to the population from nuclear waste shipments to salt domes was prepared by Bechtel.

The final Paradox regional environmental characterization report was received from Bechtel. The regional summary report and the area environmental characterization report are expected early February. An environmental evaluation of area studies in the Paradox Basin was initiated by Bechtel.

The subregional summary and area recommendation report for the Permian Region was initiated by NUS. The Environmental Evaluation for planned drilling in the Permian Basin was reviewed.

Revised proposals for regional granitic work were received from ORNL, PNL, and ANL. A funding authorization was prepared for ANL to begin regional environmental characterization in the Minnesota portion of the Lake Superior Region, and work was initiated. Work in Michigan and Wisconsin is being held up pending completion of consultation with the states.

The Commerce Business Daily announcement for region Environmental Project Managers was published, and the request for proposal was released.

WBS 1.3.7

Project: Study of Morro do Ferro

Principal Investigator: New York University Medical Center (M. Eisenbud)

ONWI Project Manager: J. O. Duguid

Objective

The objective of this project is to study the natural transport of thorium and other elements from Morro do Ferro. The thorium deposit on the Morro do Ferro, a hill on the Pocos de Caldas plateau in Minas Gerais, Brazil, offers what may be a unique opportunity to improve understanding of the behavior of plutonium over geologic time. Thorium is an actinide with chemical properties similar to plutonium in the +4 state. The Morro do Ferro deposit is estimated to contain 12,000 MT of thorium, and the deposit is believed to be about 80 million years old.

Activities During the Reporting Period

No progress report was submitted for the period.

WBS 1.3.7**Project:** Hydrology of Nearly Impermeable Rock**Principal Investigator:** Lawrence Berkeley Laboratory (T. Doe and J. Long)**ONWI Project Manager:** J. O. Duguid**Well Test****Objective**

The objective of the first task is to develop improved well test techniques for low-permeability rocks by (1) testing the constant head (pressure) injection, and (2) developing transient flow rate solutions for the constant head test for a variety of flow-path geometries by developing solutions and field methods for determining fracture apertures, the areal extent of fractures, and the nature of the hydraulic interconnection of probability (fracture boundary conditions). Such a tool can be used to generate probability density parameters of functions for fracture geometry, which in turn can be used as input to stochastic flow models and hydrologic transport risk analyses.

Progress Reported Previously

Analytical constant head, transient flow-rate well test solutions were developed for single fractures with open and closed outer boundaries. A computer data acquisition system for field reduction and analysis was assembled. Work began on the numerical modeling of constant head injection of deformable fractures.

Activities During the Reporting Period

The activities focused on (1) modeling the constant head injection of single fractures with partially open boundaries and (2) design of field test equipment.

Integrated finite difference meshes were generated for analyzing flow in partially open fractures. Specifically, the meshes simulate a single fracture and its intersection with a larger aperture fracture. The mesh size was found to exceed the limits of the existing numerical model code; however, modifications of the code are being made to handle larger meshes.

Deformable fracture modeling was done using the coupled stress-flow code, ROCMAS. A horizontal fracture of 150-meter radius with open boundaries displays the following transient flow behavior: For the first few seconds after a constant wellbore pressure is applied, the flow rate decays in a manner similar to that of a nondeformable fracture. But after this time period, the flow begins to increase, reaching steady flow after about 20 minutes. This behavior seems to reflect an opening of the fracture when the injection pressure is distributed over a sufficiently large area that the load itself is enough to cause significant deformation.

Future Activities

Modeling of transient radial flow under constant head in partially open fractures will be completed. Work on a low-flow-rate flow meter and downhole valve system for constant head well testing will be completed.

Permeability Studies

Objective

The objectives are to determine (1) when a fractured rock system may be treated as an equivalent homogeneous, anisotropic porous medium, and (2) what values of equivalent porous media properties apply to such systems. Objective (1) involves building numerical models of fracture systems and subjecting these systems to simulated flow tests. Response to these flow tests can then be compared to the theoretical response of a porous medium. When the response of the fractured system is comparable to the theoretical response, the system can be considered to have an equivalent porous media permeability. Once fracture systems are found which have equivalent media permeabilities, objective (2) will be attacked. Fracture systems will be described in statistical terms.

The object of this phase will be to find both the effect of variation of the statistical parameters and the variation in permeability between fracture systems which are different realizations of the same statistical population.

Progress Reported Previously

This task started in January, 1980. A literature survey was completed in three areas: the permeability of fractured rock systems; the theory of homogeneous, anisotropic media; and the statistics of fracture geometry. A numerical fracture system model was adopted and coupled with a fluid flow program for numerical permeability testing. The model was validated for extensive or infinite fractures, and then a random example of nonextensive fractures was studied. Criteria for behavior as an equivalent porous medium were developed.

Activities During the Reporting Period

The objectives were: (1) to add the capability of plotting isopotentials to the numerical program, and (2) to develop a method to find the "best fit" ellipse for the permeability data generated by the numerical model. Revision of the flow program was delayed since some results can be obtained without the revision.

Rather than contour isopotentials on the fracture mesh, it was decided to simply plot the locations on the fractures where the desired values of potential occurred. Work on a program to locate and plot such values was initiated this quarter. The program has been written and is now being debugged.

A literature survey was conducted to find methods for linear regression of the permeability data to an ellipse. The existing methods were evaluated and found to be inadequate. Work was initiated to evaluate the use of an iterative regression model.

Future Activities

The isopotential plot program will be completed. A regression model for "best fit" will be devised. A system of fractures which can be replaced by an equivalent porous medium will be sought.

WBS 1.3.7**Project:** Behavior of Natural Repositories**Principal Investigator:** Los Alamos Scientific Laboratory (A. E. Norris)**ONWI Project Manager:** N. J. Hubbard**Fission Product Release****Objective**

This program has been undertaken to measure both the rates of fission product release from UO_2 materials and the rates of dissolution of UO_2 matrices under controlled conditions. The major emphasis in this work is the determination of the function of oxidizing and reducing conditions in maintaining the integrity of the UO_2 matrix, with consequent retention of reactor fission products. The temperatures in these controlled conditions will be chosen to obtain data pertinent to long-term storage of UO_2 spent fuel elements. The results of the measurements obtained in this experimental program will be correlated with the observed retentions of fission products in UO_2 ores at the Oklo natural fission reactor site.

Activities During the Reporting Period

No progress report was submitted for the period.

Oklo Natural Fission Reactor Program**Objective**

Geologic burial is the favored method being considered for the disposal of commercially generated radioactive wastes. Efforts to evaluate the effectiveness of geologic media in isolating the wastes from the biosphere are limited by the necessity to consider periods of time much greater than that recorded by human experiences. This program studies the retention and migration of elements in the earth's crust by examining the geologic record which does extend over appropriate periods of time.

Activities During the Reporting Period

No progress report was submitted for the period.

WBS 1.3.7**Project:** Boundary Element Model Development—Repository Design Studies**Principal Investigator:** University of Minnesota (A. M. Starfield)**ONWI Project Manager:** M. R. Wigley**Objective**

There are two major objectives of this work. The first is to develop and apply boundary element computer models for thermomechanical analysis of repositories in bedded salt. The second is to develop constitutive models for the inelastic, time-dependent behavior of salt through laboratory investigation of its response under varying loading conditions.

Progress Reported Previously

In design studies, the boundary element solution of the benchmark problem was completed using SALTY. This solution was to be compared with that obtained using the RE/SPEC finite element code.

In code development, the complexity of the analytic solution for a heat source in a transversely isotropic material was reported, as was the difficulty in obtaining analytic solutions for displacement discontinuities within layered materials.

In the laboratory investigations, a draft report on the first phase of the salt testing program was completed, and difficulties encountered in volumeter construction led to two options for the design of the volumeter.

Activities During the Reporting Period**Design Studies**

The results obtained using SALTY for the benchmark study were written up as a separate report in October. Further analysis of the benchmark study has been shelved, pending completion of RE/SPEC's finite element study.

The task of resurrecting and documenting the REPOS Code made some progress during the reporting period. Several problems with the code were identified and rectified, and it is believed that this task is now nearing completion.

The major effort during the quarter was directed toward a sensitivity analysis of key parameters in repository design, using the SALTY code. The geometry and parameter values of the benchmark study were used as the "standard run" for the sensitivity analysis, and then each of the following parameters was varied, in most cases taking values approximately 50 percent higher and 50 percent lower than the values used in the benchmark study:

Parameters

1. Young's modulus
2. Poisson's ratio
3. Coefficient of thermal expansion
4. Thermal conductivity
5. Barrier pillar size
6. Extraction ratio
7. Thermal loading
8. Depth of repository below surface.

(It should be noted that the first four parameters are rock properties, while the next four are parameters under the control of the designer.)

The results of the sensitivity runs are being compared with specific attention to the following:

1. Room closure
2. Surface displacement
3. Tangential stresses at the surface
4. Temperatures at a distance of 90 meters above the center of the panel
5. Shear stresses 90 meters above the panel.

Additional runs will be based on the results of the initial sensitivity study. These will look at the accumulative effect of coupling two parameters, and also at the "trade-offs" between design parameters, such as between extraction ratio and thermal loading, so as to limit room closure, for example, to 10 percent.

Code Development

Work in this task concentrated on developing improved boundary element methods for thermomechanical analysis, and a topical report was completed describing the work done so far. This includes recent work on testing the behavior of Taylor kernels, and also in deriving expressions for linear variation across both the element and the time interval for a single-layer kernel.

Laboratory Investigations

Fabrication of the laboratory facility for this task is now in the final stages of construction.

SUMMARY

1.4 REPOSITORY

The scope of the Repository Task covers the development of the repository data base, development of instrumentation and equipment, design and development of repository seals, repository designs, and the performance evaluation of the repository within the total nuclear waste isolation system. The prime objective of the repository task is to develop suitable repository designs, including required instrumentation and equipment.

Heat transfer studies involving uncertainty assessment are progressing. Two reports completed this quarter deal with a simple near-field granite nuclear waste repository and the Avery Island dome salt in situ experiments. These reports will be used to determine key design performance criteria and to evaluate the uncertainties in a specific design.

Experiments to develop methods of measuring the thermal conductivity of fractured and friable materials continue. The methods will be used to obtain accurate thermal data for thermal analyses of possible repository sites.

Constant-strain-rate experiments were made, thereby completing the matrix of parameters for the experimental program of synthetic rock salt specimens. Structural characterization of the synthetic specimens and step-by-step procedures for the completed experiments were compiled. The significant result is confirmation that confining pressure, at least between 20 and 200 MPa, is not an important parameter in the control of the strength of rock salt.

Work continues to provide data for the proper design of excavated cavities in rock salt which are candidate locations for the storage and disposal of radioactive wastes. In general, the dome and bedded salt samples show remarkable similarities in their deformational behavior. These data suggest that the mechanism for deformation is the same in these two types of rock salt at moderate loads and at room temperature.

Investigations at Avery Island to acquire thermal and mechanical salt response data continue. These data will aid in the identification and understanding of the various responses of domal salt to imposed heating conditions. Specifically this quarter, the accelerated borehole closure experiments became operative and the in-mine portion of the brine movement post-test evaluation began. The extended permeability measurement test holes were drilled, and preliminary measurements were made. Failure of the data-logging system has delayed final measurements.

A major activity in the international brine migration task was preparing and issuing an updated brine migration test plan to include a firm definition of the experiments, an update of hardware development, and supporting calculations. Agreement was reached with the Asse Mine (FRG) on the basic dimensions, particularly of the source canister which must be compatible with German handling equipment.

A draft list of field test requirements for the Salt Test Facility (STF), supported by technical justification, was developed. The list reflects a mutually agreed philosophy of being limited to those tests that strictly require a facility like STF for proper performance or provide valuable data with minimal additional cost to the facility.

The Cooperative Field Tests program was instituted to coordinate various support activities in the repository data base development area. Specifically, this program is designed to (1) provide management support for field testing activities, (2) evaluate and improve existing techniques for in situ rock characterization, (3) perform experiments in southeastern New Mexico that address issues identified in the repository development programs, and (4) support the international cooperative program on nuclear waste management.

Laboratory measurements of rock behavior under elevated pressure and temperatures at simulated, deep in situ conditions continue. Sample tests run this quarter were marked by exceptionally good behavior of the experimental system and, along with the thermal properties data gathered, allowed a careful analysis of the sensitivity of the data to slight changes in temperature conditions. It is expected that these data will be correlated with field observations and later incorporated into a model for prediction.

In situ experiments in the Stripa Mine, Sweden, have yielded data on the thermomechanical response of a granite rock mass to the heat that might be produced by burial nuclear waste. The materials properties test program continues to study and acquire thermomechanical properties using drill core (small and ultra-large) from Stripa. Because of the volume of data, the results to date should be viewed as preliminary. The near-field underground stress measurements will play a key role in validating the mechanical model of the Stripa tests.

In situ heater tests, hydrologic tests, and fracture system characterization studies in granite are being conducted in Stripa. These studies will help assure long-term containment of potential contaminants during the underground storage of nuclear waste in crystalline rock. Compilation of data on a hole-by-hole basis has begun. The data will furnish a ready reference for calibrations, instrument down-time, instrument maintenance, and quirks in data that can be traced to operational causes.

The international Stripa testing program was initiated during this quarter. This program will include hydrogeological investigations in boreholes, migration in single fissure (preparatory study), and a buffer mass test.

The Stripa data analysis program will allow better understanding of the response of a fractured granite rock mass to thermomechanical loading. Efforts this quarter were directed toward examination of temperature data to determine which thermocouples provide: (1) reliable data to be compared with numerical models, (2) estimation of the effect of fractures on thermal conductivity of the rock mass, and (3) analysis of the method of data interpretation for USBM borehole deformation gages.

All activities in the expected repository environments in granite (ERE-G) program during this quarter were related directly to participation in the Reference Repository Conditions Interface Working Group (RRC-IWG), and no further technical activity was undertaken. Activities in support of the RRC-IWG included submission of Interim Reference Repository Conditions for Commercial and Defense High-Level Nuclear Waste and Spent Fuel Repositories in Granite to the RRC-IWG, which approved this draft and submitted it to ONWI management review. Drafts of similar reports for salt, shale, and tuff were also reviewed as a part of ERE activities.

The heated block tests in granitic gneiss are used to determine the in situ mechanical, thermal, and hydrological properties of a jointed crystalline rock under conditions of varying

temperature and stress. The main activity this quarter was to perform three different load-path cycles under ambient (12 C) conditions. The resultant effects of joint closure, opening, and shear are clearly reflected in the results.

The NWTS repository equipment development plan program is a comprehensive testing effort to ensure that the waste handling equipment installed at the repository will function according to predicted availability goals. Work this quarter resulted in the preparation of a draft repository equipment test plan and draft statements of work for equipment technology development.

The engineering design criteria methodology development study was completed this quarter. The study developed a methodology and format which will be used to aid in preparing engineering design criteria and specifications for equipment used at a nuclear disposal facility.

The main thrust of the equipment reliability, availability, and maintainability (RAM) program was completion of the RAM Guidebook. The two-volume guidebook will represent the overall RAM methodology recommended for the NWTS program.

Development of a single-hole directional electromagnetic antenna has progressed, with measurement of the antenna radiation pattern in granite at Gold Hill, Colorado.

A prototype long-wavelength system to transmit sensor data collected within a subsurface nuclear waste repository without breaching the geologic integrity was examined.

A preliminary design concept in the form of a working design for sealing a repository at the Los Medanos site was developed during this quarter. The design is intended to stimulate comment and thought and will be revised periodically until the conceptual sealing design for the site has been completed in August, 1982.

Negotiations have been completed for the design, construction, and laboratory testing of the fluid permeability measurement system for the borehole plugging program. Once the system has been adequately tested in the laboratory, plans for a full-scale in situ evaluation will be initiated.

The seal materials performance limits and longevity program began work on evaluating the nine mixtures made in optimizing the Bell Canyon Grout, BCT-1ff. Laboratory-cast and field-cast grouts from the proportioning studies, ERDA-10 plugs, and Bell Canyon plugs underwent physical tests and examinations by SEM and X-ray diffraction. Their properties reflected developments from earlier tests. This program will optimize proportions for grouts to be used in tests at AEC-7 and similar geologic settings.

The program to obtain experimental data on the wellbore damage zone caused by full-scale drilling operations has been extended to include five rock types: anhydrite, salt, granite, basalt, and tuff. These additions were made to obtain fundamental data for the design of sealing systems in each host rock types. Tests conducted on anhydrites confirmed the data that were collected in the field at the Bell Canyon Test site. The tests have conclusively shown that all flow has occurred around the plug, i.e., between the cement plug's outer wall and the side wall of the hole. Consequently, we now have an established method for testing sealing designs in the laboratory.

All NWTS Conceptual Reference Repository Description (CRRD) systems affected by horizontal burial were reviewed. The comparison of horizontal versus vertical emplacement for host rock media other than salt will be made on both a technical and an economic basis.

Fabrication of the major components of the salt core holder for the salt core brine migration experiment was completed. It is planned to consolidate large particle size crystalline salt in brine in the initial experiments to shake down equipment and to extend the consolidation experiments to 10-cm-diameter columns.

The impact of defense waste on repository design project focuses on determining the impacts of receipt and emplacement of high-level and transuranic defense wastes on existing NWTS geologic repository designs. The work performed this quarter included: (1) completing the defense waste data collection, (2) developing scenarios for the distribution and allocation of defense wastes to national and regional repositories, (3) developing repository characterizations from existing conceptual designs, and (4) identifying the top-level repository impacts caused by the receipt and emplacement of defense waste.

During this quarter, the codes SPECTROM and STEALTH were compared using a room-scale benchmark problem. The SPECTROM programs continue to be developed and verified as required to provide state-of-the-art analysis capability for rock mechanics problems unique to the ONWI program.

During this quarter, significant progress was made in the convective heat transfer in room and pillar geometries project in three areas: (1) continued accumulation of data for the smooth-walled case, (2) development of a computer program for data analysis, and (3) linear regression analysis for the development of a preliminary correlation for asymptotic Nusselt numbers in the fully developed turbulent flow regime. Heat transfer data are needed to assess ventilation and cooling requirements in the repository during waste canister emplacement and retrieval to ensure worker safety.

The problem of determining the temperature distribution within a geologic medium which is driven by a heat source due to contained nuclear waste has been investigated analytically and numerically. Approximate temperature profiles will be calculated and incorporated into the repository model.

The development of the thermo-mechanical-hydrological (TMH) fractured rock code ROCMAS continued. This code will be used to develop a model needed to predict the effects of waste heat and to assess the movements of ground water and possible radionuclides between the repository and the biosphere.

A project for the analysis of thermal-mechanical-hydrological response of rock was initiated. This project will further develop and verify the STEALTH computer code as it applies to geologic isolation of nuclear wastes. The project will also perform generic design studies that will form the design and licensing data base for site-specific analyses.

WBS 1.4.2

Project: ONWI Support Program: Program Coordination and Special Problems

Principal Investigator: Oak Ridge National Laboratory (A. L. Lotts)

ONWI Project Manager: G. E. Raines

Heat Transfer Studies

Objective

The ultimate objective for the subtask involving uncertainty assessment is a general methodology that can be used to determine the modeling parameters and physical data that are most important in determining key design performance criteria and to obtain the uncertainties in a specific design.

Progress Reported Previously

A report entitled *A Statistical Sensitivity Analysis of a Simple Nuclear Waste Repository Model* was published. The Avery Island calculations had not been previously published.

Activities During the Reporting Period

In the subtask involving uncertainty assessment, the results for a repository were included in a draft report, *Sensitivity and Uncertainty Analysis of a Simple Near Field Granite Nuclear Waste Repository*, which has been submitted for review.

A draft entitled *Thermal Analysis Supporting the Design of the Avery Island Dome Salt In Situ Experiments* has been prepared and is undergoing internal editing.

Future Activities

Both reports mentioned above will be put in final form and published.

Expected Repository Environments

Objective

The objective is to identify the expected environments for nuclear waste and spent fuel repositories in bedded and in domal salt. The environments or effects include temperature distributions and histories, brine flows, brine chemistry, and pressure.

Progress Reported Previously

The report on commercial waste and spent fuel, *Expected Environments in High-Level Nuclear Waste and Spent Fuel Repositories in Salt*, was issued at the end of the previous quarter.

Expected environments for a defense high-level waste (DHLW) repository in salt have been predicted analogously to previous analyses for spent fuel and reprocessed commercial high-level wastes (CHLW). Environments predicted include near-field and far-field temperatures, fluid pressure, and nuclear radiation fields. Some sensitivity studies have also been performed. The main results of the calculations include the following:

1. Rock temperatures do not exceed 86 C. Canister wall temperatures and waste temperatures are less than 94 C and 99 C, respectively.
2. The maximum brine inflow rate to an emplacement hole is 0.015 liters/year, occurring in the first 30 years after emplacement. The total accumulation of brine migrating to the emplacement hole after 1,000 years is less than 0.5 liter.
3. Gas pressures encountered by the waste package do not exceed 0.36 MPa prior to mine closure. After this time it is conceivable that stress on the canister could approach the lithostatic rock stresses.
4. Maximum dose rates in the salt are less than 1,400 rads/hour.

Activities During the Reporting Period

The review comments were considered, and a new draft was prepared that is being made ready for publication.

Future Activities

The final draft of *Expected Environments for a Defense High-Level Waste Repository in Salt* will be completed.

WBS 1.4.2

Project: ONWI Support Program: Thermal Property Measurements

Principal Investigator: Oak Ridge National Laboratory (J. G. Moore and M. T. Morgan)

ONWI Project Manager: G. E. Raines

Objective

The objective of this program is to obtain accurate thermal data to be used in thermal analysis of possible repository sites. Methods and procedures will be established for making thermal property measurements, and data will be obtained on selected specimens of cores from possible repository sites and standard natural materials.

Progress Reported Previously

Final thermal conductivity measurements were made on the Pyroceram standard using graphite cloth at the interfaces to improve thermal contact. However, the graphite cloth increased the temperature drops across the interfaces by a factor of ~ 3 , thus increasing thermal conductivity measurements by 6 percent.

The pressure applied to the 5-cm-diameter area was approximately 4 kg. If sufficient pressure was applied to crush the graphite, better results might be obtained. Lead foil and nickel powder in silicone grease were tried in previous experiments. The lead foil also increased the temperature drops across the interfaces, while the nickel powder had no effect. These methods might be effective where surface roughness was excessive; otherwise the most accurate results can be obtained by careful surface preparation.

A series of thermal conductivity measurements was initiated to test a thermal conductivity cell in which a fragmented or friable sample is immersed in a liquid. The first measurements were made on water alone to determine the characteristics of the cell. The values obtained were within 4 percent of the handbook value. Measurements were started on Pyroceram and glass.

Activities During the Reporting Period

Experiments to develop methods of measuring the thermal conductivity of fractured and friable materials are continuing. Two methods are being considered: One is a conductivity cell in which the samples will be surrounded with a fluid phase, and the other is the use of coatings to improve the thermal conductivity at the interfaces of samples which cannot be prepared properly due to fragmentation or pitting.

This cell consists of a hollow, thin-walled plastic cylinder capped with metal ends. The sample is placed in the open cell, the air evacuated, and the sample and cell flooded with water. The cell is then sealed with the metal caps. The cell assembly is then inserted into the cut-bar apparatus for thermal conductivity measurements. Since the measurements are made at atmospheric pressure, the temperatures are limited to less than 100 C.

The equations used to determine the thermal conductivity of the aggregate sample in the cell, based on basic electrical conductivity equations, are:

1. Series conductivity:

$$k_s = \frac{k_a(1-\Phi)}{1-(\Phi k_a/k_w)}$$

2. Parallel conductivity:

$$k_s = \frac{k_a - \Phi k_w}{1-\Phi}$$

3. Aggregate conductivity (derived from Equations 1 and 2):

$$k_a = \frac{D^2}{d^2} k_c - \frac{(D^2 - d^2)}{d^2} k_p$$

where:

k = thermal conductivity (W/m·K)
 D = outer diameter of cell (cm)
 d = inner diameter of cell (cm)
 Φ = porosity

and the subscripts are:

a = aggregate
 c = complete cell
 p = plastic wall
 s = solid sample
 w = water.

In these studies, glass was selected to serve as a model to test the cell and to determine the effect of sample fracturing on the thermal conductivity. Plate glass samples 4.6 cm in diameter and 5.1 cm in diameter by 1.27 cm thick were prepared. The thermal conductivity was determined before and after fracturing. Fracturing was accomplished by heating the glass to 170 C and dropping it in ice water. The porosity induced was 0.08 percent. The fractures extended in all directions and produced grains approximately 0.6 cm across. Thermal conductivities determined for glass at 40 C are listed in Table 1.

TABLE 1. THERMAL CONDUCTIVITIES OF PYREX GLASS

Pyrex Glass	Thermal Conductivity, W/m·K	
	Measured	Calculated
Intact disc	0.98 to 0.99	
Cracked disc	0.92 to 0.98	
Intact disc in cell	0.77 to 0.78	0.91
Cracked disc in cell	0.77 to 0.78	0.91

No corrections were made for porosity introduced by cracking the glass. The thermal conductivity of glass measured in the cell is 1 to 7 percent less than that measured directly. No significant change was caused by cracking. Using Maxwell's dispersion equation (1891) as rearranged by Beck (1976), the change in thermal conductivity for the 0.08 percent of porosity induced by cracking would be 1.5 percent, which is within the range of experimental uncertainty (5 percent). The Maxwell equation is:

$$k_a = k_s \frac{(2r + 1) - 2\Phi (r-1)}{(2r + 1) + \Phi (r-1)}, \text{ where } r = k_s/k_w$$

In the second method, various coatings are being examined as possible agents for improving the thermal contact between interfaces with uneven or pitted surfaces. The coating should adhere well to the sample surface, produce no voids, have a low enough viscosity to allow spreading without damage to a friable surface, and match the thermal conductivity of the sample. Glass samples with ideal surfaces and previously measured conductivities are being used as bases for the determination of the effect of the coatings. Tests are in progress with five different epoxy resins, two of which are aluminum filled. The others will be examined with and without various concentrations of nickel metal powder filler. Initial tests showed that the presence of nickel powder produced no change in the thermal conductivity value obtained with a glass specimen that had a flat surface. The tests will be repeated with specimens having pitted, uneven surfaces such as would be expected with friable materials which are difficult to prepare. The resins have also been cast into regular cylinders to obtain their thermal conductivity. The first unfilled epoxy resin had a thermal conductivity of 0.52 to 0.54 W/m·K at 40 to 75 C.

Future Activities

Work will continue on the method of measuring the thermal conductivity of fragmented and friable materials.

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WBS 1.4.2

Project: Bench-Scale Salt Creep Tests, Phase III

Principal Investigator: South Dakota School of Mines and Technology (C. A. Kliche)

ONWI Project Manager: M. M. Lemcoe

Objective

The objective of this program is to provide bench-scale, time-dependent deformation data on samples of dome-type rock salt. These data will be used by other research groups to check existing computer codes on models that predict stress-induced deformation.

A method for measuring deformation of rock salt using photoelastic stress-strain measuring techniques and brittle coating techniques is to be developed.

Progress Reported Previously

1. Laboratory scale testing of stresses in rock salt using brittle coating and photoelastic measurement techniques has been done. Results of this phase indicated some correlation between strain gage and photoelastic testing, but additional fine tuning is required to increase the level of confidence in the results.
2. Brittle coatings are useful for testing machine parts and other metal or plastic structures, but they are not suited for rock salt testing. Also, bonding of the birefringent material to salt over long periods of time does not seem to be a problem.
3. Fabrication, assembly, and inspection of the biaxial load frame has been completed. The frame was pressure-tested to assure that no undue bending would occur.

Activities During the Reporting Period

Activities during the reporting period have been aimed at completing the uniaxial tests. The final test has not yet begun, so no results are available.

Work during the last quarter was also done on the first of the salt blocks to be used for the biaxial test. Work on the biaxial block has been limited mainly to obtaining a finish on the block.

Future Activities

The first priority for next quarter is to complete the uniaxial testing for time-response creep data using photoelastic methods. Once the uniaxial test is running, three more objectives are scheduled for next quarter: (1) finish the biaxial samples, (2) test one of the samples in the biaxial load frame, and (3) begin the laboratory scale photoelastic testing of salt and one other natural material (to be selected, based on material availability).

WBS 1.4.2

Project: Transient Creep in Rock Salt

Principal Investigator: Texas A&M Research Foundation (J. Handin)

ONWI Project Manager: M. M. Lemcoe

Objective

The objectives are to (1) develop a transient creep constitutive relation for rock salt based on operative deformation mechanisms, (2) measure, experimentally, the state variables required for evaluating transient creep theories, and (3) determine the deformation mechanisms during transient and steady-state creep of rock salt.

Activities During the Reporting Period

Overview

Constant-strain-rate experiments at 10^{-4}s^{-1} , 20-MPa confining pressure and temperatures at 100, 200, 300, and 390 C were made, thereby completing the matrix of parameters for the experimental program on the synthetic rock salt specimens. Structural characterization of the synthetic specimens, including selected ones that had been stress-relaxed, continued. Step-by-step procedures for the recently completed constant-strain-rate experiments on the synthetic specimens were compiled.

Constant-Strain-Rate Experiments

Figure 1 shows the data and results for constant-strain-rate experiments on synthetic specimens at 10^{-4}s^{-1} , 100, 200, and 300 C, and 50, 100, and 200-MPa confining pressures. This matrix of parameters was completed by deforming the synthetic specimens at 20-MPa confining pressure and the three temperatures (Figure 2). Reproducibility continues to be good, particularly in the steady-state region (around 4 to 10 percent strain), and the influence of temperature is to weaken, as expected. When only data for three confining pressures were available, the influence of confining pressure was considered very small. The data at 200 C have been used to point out that the average strength at 10 percent strain increases from 11.6 to 12.6 to 13.7 MPa as the confining pressure increases from 50 to 100 to 200 MPa. The average strength of specimens deformed at 20 MPa and 200 C is 11.7 MPa, that is, the same as the 50 MPa. Clearly the confining pressure effect is negligible.

Structural Characterization

Optical study of thin sections of the constant-strain-rate experiments has continued, with examination of stress-relaxed specimens and those deformed at 20-MPa confining pressure. Early results indicate the stress-relaxed specimens are no different from their non-relaxed counterparts. For example, specimen 812 (100 C, 100-MPa confining pressure, $\Delta\sigma =$

FIGURE 1. STRESS-SHORTENING CURVES FOR SYNTHETIC SPECIMENS OF PURE ROCK SALT SHOW EFFECTS OF TEMPERATURE AND PRESSURE ON THE STRENGTH AT A CONSTANT STRAIN RATE OF $10^{-4}s^{-1}$

Specimen numbers are indicated for each curve. Dashed curves are for tests at 200-MPa confining pressure; solid curves are for those at 100 MPa; and dotted ones are for tests at 50 MPa.

FIGURE 2. STRESS-SHORTENING CURVES FOR SYNTHETIC SPECIMENS OF PURE ROCK SALT SHOW EFFECTS OF TEMPERATURE AT A CONSTANT STRAIN RATE OF $10^{-4}s^{-1}$ AND 20 MPa CONFINING PRESSURE

$\sigma_1 - \sigma_3 = 26$ MPa at 10 percent shortening), which was not stress-relaxed, is essentially identical with regard to grain size, axial ratio, cubic inclusions, polygonization, etc., to specimen 822 relaxed to $\Delta\sigma = 15.7$ MPa after 1,800 seconds. In addition, initial cursory examination of specimens deformed at 20-MPa confining pressure show no conspicuous differences compared with those at a higher confining pressure.

Work has begun on jacketing and strain-gaging the large specimens of natural rock salt.

The step-by-step procedure for preparing synthetic specimens is as follows:

Compaction Vessel (CV) Billet Production

1. 320 g of granular pure NaCl is poured into a 3.8-cm diameter polyolefin heat shrink tube.
2. The polyolefin is sealed to the upper and lower pistons of the CV.
3. The specimen is evacuated through a port in the upper piston under modest vacuum.
4. The specimen is hydrostatically preloaded (uniform confining pressure) to about 17 MPa.
5. The strip heaters, located on the outside of the CV, are turned on, and the temperature is equilibrated (20 minutes) at 100 C.
6. Confining pressure is increased and equilibrated at 150 MPa for 15 minutes.
7. The strip heaters are turned off.
8. The confining pressure is removed.
9. The specimen is allowed to cool about one hour in the CV before it is removed and placed in a desiccator.

Constant-Strain-Rate Testing in Variable-Strain-Rate (VSR) Apparatus

The step-by-step procedure for performing constant-strain-rate experiments is as follows:

1. Right-circular cylinders 2.5 cm in diameter by 4.8 to 5.7 cm long are cored from the billets described above. The ends of the cylinders are machined with a surface grinder.
2. Specimens are stored in a desiccator prior to the experiment.
3. The specimen is jacketed in a 10 to 20-mil lead sleeve and placed in the pressure vessel.
4. The specimen is uniaxially preloaded to about 1.3 MPa.

5. Confining pressure is applied to achieve a hydrostatic stress rate.
6. The furnace is turned on and temperature equilibration begins.
7. After the temperature has equilibrated (45 minutes at 300 C) the confining pressure is increased.
8. The specimen is now ready to be deformed by increasing the differential stress at the fixed strain rate.

Significant Results

The most significant result is the confirmation that confining pressure—at least between 20 and 200 MPA—is not an important parameter in the control of the strength of rock salt. This is reflected in the stress-shortening curves, which show temperature to be far more influential, and in the thin-section structural characterization work.

Future Activities

During the next report period, the following will be done: (1) assembly of new large-specimen apparatus, (2) jacketing and strain gaging the 10 x 20-cm cylindrical specimens for the new apparatus, (3) applying the curve-fitting methods to analyze the recently completed constant-strain-rate-test results, and (4) structural characterization of (a) the experimentally deformed synthetic specimens, (b) the natural rock salt to be used in the new large-specimen apparatus, and (c) the experimentally deformed large specimens.

WBS 1.4.2

Project: Salt Model Pillar Studies

Principal Investigator: Oak Ridge National Laboratory (T. F. Lomenick)

ONWI Project Manager: G. Raines

Objective

The objective of this work is to provide data for the proper design of excavated cavities in rock salt which are candidates for the storage and disposal of radioactive wastes.

Activities During the Reporting Period

Model pillar tests are proceeding under the conditions outlined in Table 1 on cores of rock salt from the domes listed and the Wellington Formation.

Preliminary data from the Cypress Creek Dome and the Wellington Formation are summarized in Tables 2 and 3.

In general, the two samples show remarkable similarities in their deformational behavior. For both the dome salt as well as the bedded deposit there is initially a high ratio of deformation that continues to decrease with time even after 700 hours of loading. These data suggest that the mechanism for deformation is the same in these two types of rock salt at moderate loads and at room temperature.

All samples are prepared in the same manner, aiming to simulate pillar, roof, and floor conditions that would exist in mined cavities in rock salt deposits. The test specimens used in

TABLE 1. TEST CONDITIONS

Sample	Compressive Load, kg/cm ²	Temperature, C	Duration, days
1. Cypress Creek Dome, MS	280	22.5	90
2. Cypress Creek Dome, MS	420	60.0	90
3. Rayburn's Dome, LA	280	22.5	30
4. Rayburn's Dome, LA	420	60.0	90
5. Wellington Formation, KS	280	22.5	30
6. Wellington Formation, KS	420	60.0	90

**TABLE 2. PRELIMINARY DATA FROM
CYPRESS CREEK DOME**

Temperature, 22.5 C;
Compressive Load, 280 kg/cm²

Time, hours	Vertical Shortening of Pillar, percent
1	0.86
5	1.25
10	1.63
80	2.47
145	2.68
196	3.12
311	3.35
359	3.67
407	3.87
527	4.23
647	4.40
700	4.62

this work are cylindrical in shape, with a portion of the center ground out to form the pillar and surrounding rooms. By epoxying steel rings around the ends of the samples, effective confining pressure is applied to the roof and floor portions of the models when they are loaded. Constant uniaxial loads are applied to the models by hydraulic compression testers having capacities up to 136,000 kg (300,000 pounds). Cavity closure is measured by mounting two dial gages 180 degrees apart on the rings. Elevated temperature tests are performed with the specimens inside a cylindrical heating jacket and with barrier heaters on top and bottom between the specimen and the platens.

Most tests are performed on model pillars fabricated from 8.73-cm (3.44-inch) diameter cores of rock salt. For model preparation, all cores were first cut to their approximate desired length by handsaw and then finished by machine sanding. Next, steel confining rings (ASTM A-108, Grade 1018) are affixed to the tops and bottoms of the cores using Ciba Araldite 502 epoxy and Ciba hardener 951 for samples heated up to 100 C, Ciba Araldite 6005 epoxy and Pyromellitic Dianhydride (PMDA) hardener are used. The 1.5-cm (0.6-inch) high unconfined center portion of the sample is then ground out, using a sanding disc attached to a standard drill press, to form the size pillar desired. Sheet Teflon greased with a mixture of silicon grease and graphite is inserted between the tops and bottoms of the samples and the platens of the compression machines to reduce the friction between the salt and the platens to an insignificant amount.

TABLE 3. PRELIMINARY DATA FROM THE WELLINGTON FORMATION

Temperature, 22.5 C;
Compressive Load, 280 kg/cm²

Time, hours	Vertical Shortening of Pillar, percent
1	0.67
5	1.05
9	1.40
79	2.37
146	2.80
194	3.25
310	3.55
357	3.88
406	4.08
526	4.57
646	4.67
699	4.92

To ensure a uniform temperature within the heated model specimens, Chromel Alumel thermocouples are emplaced in 1.5-mm (0.06-inch) diameter holes drilled into the center of the pillar, as well as at the outer edge. A 12-point strip-chart recorder with a 0 to 350 C range is used to trace the temperature curves. A constant voltage transformer is used in supplying 118 volt AC to adjustable autotransformers which are used as temperature controls for the cylindrical heating jackets, as well as the heaters in the platens at the top and bottom of the sample. In this manner, temperatures are regulated to within 1 to 2 percent of the desired values.

In general, the same test procedure is used for all specimens to ensure uniformity. This procedure consists of rapidly applying a compressive load corresponding to 75 percent of the testing load and releasing it and then applying a load of 85 percent of the test load which is also immediately released. A base reading for the dial gages is then taken. These initial loadings serve to seat the dial gages, used to measure deformation, and to set the platens firmly on the sample. The full test load is then applied within a few seconds to the specimen and deformation is recorded, first at intervals of seconds, then to minutes, and then at progressively longer intervals.

Future Activities

Tests will continue on model salt pillars.

WBS 1.4.2

Project: Strain-Related Radiation Damage Measurements in Rock Salt for Waste Disposal Applications

Principal Investigator: Brookhaven National Laboratory (P. W. Levy)

ONWI Project Manager: J. F. Kircher

Objective

The aspects of radiation damage in both natural rock salt and synthetic NaCl crystals and in granite, basalt, and other minerals which will be important for selection of a radioactive waste repository site, site design and engineering, evaluation of interactions between waste and host rock, risk assessment, and other considerations are being investigated. Unique equipment at Brookhaven National Laboratory is being used to make optical and other measurements on samples before, during, and after irradiation with 0.5 to 3.0-MeV electrons. The ability to make measurements during irradiation is particularly important. Rock salt and other minerals surrounding emplaced waste will be continuously irradiated, and it is essential to determine whether important radiation damage processes occur during irradiation which might not be detected in samples studied only after irradiation. The principal means of studying radiation damage used up to the present time is the determination of F-center, i.e., negative ion defect, concentrations, and the concentrations and size of sodium metal colloid particles. The formation of these and other defects during irradiation and annealing, and other effects occurring after irradiation, are being studied as a function of dose rate, total dose, sample temperature during irradiation, strain applied prior to and during irradiation, etc. Also, synthetic NaCl and natural rock salt samples from different geologic locations, including some potential repository sites, are being studied. The goal of the research is to provide sufficient information, as analytical expressions, graphs, tables, etc., to determine the radiation damage developed in rock salt, granite, basalt, and other minerals surrounding emplaced canisters as a function of radiation levels, time after emplacement, mineral temperature, and other parameters. To extrapolate from the laboratory studies to the dose rates, exposure times, and other parameters applying to repositories, it will be necessary to make a reasonably complete determination of radiation-damage formation and annealing kinetics. Also, it will be necessary to determine whether radiation damage in rock salt from different localities is similar or whether a detailed study must be made for each locality under consideration. Approximately half of the current studies are being made on rock salt, but granite and basalt have been phased into the program.

It is now clearly established that radiation damage formation in both natural and synthetic rock salt is strongly dependent on strain. For example, the colloid-growth induction period decreases as strain applied prior to irradiation is increased. The strain data strongly indicate that damage formation, particularly the colloid-nucleation process, is related to strain-induced dislocations. Consequently, a temporary theoretical effort is under way to determine which dislocation-related effects are important for radiation damage processes and are most likely to induce nucleation sites for colloid particles.

Activities During the Reporting Period

No progress report was submitted for the period.

WBS 1.4.2

Project: Radiation Chemistry of Salt Mine Brines and Hydrates

Principal Investigator: Oak Ridge National Laboratory (G. H. Jenks)

ONWI Project Manager: J. F. Kircher

Objective

The objective of this program is to experimentally establish the value for $G(H_2)$ and G (accompanying oxidants) in salt mine brines and hydrates ($MgCl_2$ hydrates) under gamma irradiation at temperatures up to about 450 K, for some solutions for which the vapor pressure does not exceed about 0.3 MPa at the test temperature. The G -value is a measure of product yield during radiolysis. It is defined as molecules of product over 100 eV of energy absorbed by the system. The radiolytic products from brine irradiation may have deleterious effects on package components via hydrogen embrittlement or enhanced corrosion.

Progress Reported Previously

Experimental methods, equipment, and procedures were developed. A large number (27) of experimental measurements of $G(H_2)$ in NaCl-saturated $MgCl_2$ solutions were completed.

Methods of analysis for minute amounts of Cl_2 and H_2O_2 in the irradiated brines were also developed. Investigation of possible analysis methods for ClO_3^- continued.

A topical report of the work was in preparation.

Activities During the Reporting Period

A draft of the topical report of the work during FY 80 was completed and submitted for review.

Future Activities

The draft report will be revised to accommodate reviewer suggestions.

WBS 1.4.2

Project: Long-Term Monitoring and Analysis of the Avery Island Heater Experiments

Principal Investigator: RE/SPEC, Inc. (L. L. Van Sambeek)

ONWI Project Manager: W. F. Ubbes

Objective

The objective of the investigations at Avery Island is to acquire thermal and mechanical salt response data that will aid in the identification and understanding of the various responses of domal salt to imposed heating conditions. The transfer of heat into the salt formation results in increased temperatures which induce thermomechanical and possibly thermochemical responses. The data obtained during these investigations will be used to either develop models describing the various responses or test existing models and assess their capabilities for describing the responses.

Progress Reported Previously

RE/SPEC has installed and is operating three in situ heater tests in the Avery Island salt mine. The tests were in operation by 1 June 1978. The design of the brine movement experiments began in February, 1979; the experiments were installed during the summer of 1979 and were in operation by 1 October 1979. One of the brine movement experiments is now in a post-test evaluation stage. Specifications and designs for the additional studies have been developed, and the installations are being made at this time. The duration of the additional studies will carry over into FY 82.

Activities During the Reporting Period

Objectives

The primary objective during the quarter was to bring the accelerated borehole closure experiments into operation. Secondary objectives were to submit the displacement report for the three heater tests, to complete the in-mine portion of the brine movement post-test evaluation, and to perform the extended permeability measurements.

Results

Heater Tests. The three heater tests, in operation for about 1,000 days, have all reached a "steady state" condition. Work during the quarter was concentrated on continuing data analysis and planning for the post-test evaluation of the Site A and Site B heater experiments. Review comments were received concerning the Engineering Test Plan (ETP) for the post-test evaluation of Sites A and B. These comments will be incorporated, as appropriate, into the final ETP before the heater tests are shut down. The post-test evaluation of the Site A and Site B heater tests is now scheduled to begin before mid-February, 1981.

Brine Movement Studies. Several of the desiccant cartridges used in the Sites NB (natural brine) and SB (synthetic brine) moisture collection systems have been analyzed for deuterium oxide concentration by the USGS. Deuterium oxide was the tagging substance used in the synthetic brine emplaced in the Site SB experiment. The concentration of deuterium oxide in the emplaced brine was 80,000 molecules per 10^6 molecules of H_2O . Significant quantities of deuterium oxide were detected in the Site SB desiccant cartridges, but these are not really indicative of the deuterium oxide concentrations in the heater boreholes. A summary of the concentration analysis for the Sites NB and SB desiccant cartridges is given in Table 1. The concentration of deuterium oxide in standard mean ocean water (SMOW) is 316 molecules per 10^6 molecules of H_2O . Hence, as shown in Table 1, the concentrations of deuterium oxide in the Site SB cartridges are significantly higher than SMOW, while the concentrations found in the Site NB cartridges were lower. Additionally, the Site NB concentrations are similar to those found in the brine pools near the test area.

From the foregoing discussion, it would appear that the synthetic tagged brine emplaced in the Site SB experiment did reach the heater borehole. The mechanism or series of mechanisms through which the synthetic brine reached the heater borehole is not presently known; however, the post-test analysis of the Site SB overcore could explain these mechanisms.

Additional Studies. Two accelerated borehole closure experiments are currently in operation at Avery Island. One test is being operated at ambient mine temperature, and the other is at an elevated temperature of 60 C. Both tests are being step-loaded in 250-psi (1.75-MPa) increments. The elevated temperature test is currently at a pressure of 750-psi (5.17 MPa), while the ambient test is at 1,000-psi. The average diametrical closure at 1,000-psi for this test as a function of time is presented in Figure 1. A linear least squares approximation of the data is

TABLE 1. DEUTERIUM OXIDE CONCENTRATIONS IN SELECTED DESICCANT CARTRIDGES FROM THE AVERY ISLAND BRINE MIGRATION EXPERIMENT MOISTURE COLLECTION SYSTEMS

Desiccant Cartridge and Site	Collection Period	Approximate Deuterium Oxide Concentration, molecules/ 10^6 molecules H_2O
NB-0	1/21/80–1/29/80	~78
NB-1	1/29/80–5/7/80	~22
NB-2	1/29/80–5/7/80	~1.8
NB-3	1/29/80–5/7/80	Insignificant
SB-1	1/29/80–5/7/80	Greater than 632(a)
SB-2	1/29/80–5/7/80	Greater than 632(a)
SB-3	1/29/80–5/7/80	Not Determined
SB-1	9/2/80–9/4/80	~15,000
SB-2	9/2/80–9/4/80	~6,670
SB-3	9/2/80–9/4/80	Insignificant

(a) The measuring system swamped the filament; therefore, the values indicated are only approximate minimum values of deuterium oxide concentration in the desiccant cartridges.

FIGURE 1. DIAMETRICAL CLOSURE AS A FUNCTION OF TIME FOR THE 1,000-psi LOAD STEP OF THE AMBIENT TEMPERATURE ACCELERATED BOREHOLE CLOSURE EXPERIMENT

also shown in the figure. The accelerated borehole closure studies are being operated under loading procedures that are significantly different than those specified in the ETP. Basically, the flatjack pressurization schedule was modified from a single-step loading to that of incremental loadings of 250-psi (1.75 MPa), followed by monitoring of the closure after each loading step. The change complicates the analysis of the data, but the study still lends itself to numerical simulation.

During the quarter the test holes for the extended permeability measurements were drilled, and preliminary falling-head permeability measurements were made in each of the holes. However, failure of the data-logging system has delayed the final measurements.

Future Activities

The following goals have been established for next quarter.

1. Complete the extended permeability measurements.
2. Drill several 12-inch-diameter cores for use in the RE/SPEC laboratory.
3. Drill five 16-inch-diameter cores for use by Terra Tek.
4. Finalize a flatjack design and install the phase II borehole closure experiments.
5. Begin the post-test evaluation of the Site A and Site B heater tests.
6. Have draft final reports submitted for the Site A and Site B heater tests.
7. Prepare an Engineering Test Plan for instrumentation of the slope from the 880-foot level of the mine to the 110-foot level.
8. Discuss and reach tentative agreement with the International Salt Company about a possible expansion of the present test area or development of a new test area on the 880-foot level.

WBS 1.4.2

Project: Experiment Development to Support Field Testing

Principal Investigators: Westinghouse Advanced Energy Systems Division (AESD)
(G. R. Kilp and F. F. Klein) and D'Appolonia (L. Jarolimek)

ONWI Project Manager: M. W. Anthony

Objective

The principal objective is to develop test plans and hardware for future field tests to study (1) the phenomenon of brine migration in salt under simulated nuclear waste repository conditions, (2) waste-rock interactions under conditions simulating long-term barrier and waste degradation, and (3) the performance of engineered barriers under geological conditions.

Progress Reported Previously

Work on this project began in December, 1979, with effort directed toward two tasks: Brine Migration and Waste-Rock Interaction. There is also a third task currently active (Technical Assistance).

Brine Migration

The first issue of the test plan, in May, 1980, reflected the results of meetings with both FRG and U.S. personnel and incorporated many features to resolve the interactive effects of many competing conditions: temperature level, temperature gradient, radiation effects, and peripheral interface conditions at the geological boundary. In September, 1980, at a joint meeting, it was decided to proceed with a Revised Brine Migration Test Plan. The most significant outcomes of that meeting were (1) recognition of the lack of liquid water inclusion in the salt at Asse, which will preclude testing for liquid inclusion migration; (2) determination that the radioactive source-handling equipment has not yet been defined, and (3) preliminary identification of a test site in the Asse mine.

Waste-Rock Interaction

Based on extensive reviews of pertinent literature, inputs from ONWI, and personal contacts with other key technical organizations, the preliminary concepts of a test plan for waste-rock interaction testing at the Asse mine were written. The test plan concepts were subjected to semiformal peer reviews before representatives of ONWI, DOE, cognizant U.S. technical organizations, and the FRG. Concerns were raised over the basic issues and objectives to be addressed by waste-rock interaction tests in salt. Resolution of the latter problem area to the satisfaction of all concerned parties was found to be crucial to the progress of the task.

Technical Assistance

The Brine Migration Test Program being conducted at Avery Island was reviewed, and a Post-Test Evaluation Program Plan was written.

Activities During the Reporting Period

Brine Migration

The major activity was preparing and issuing an updated version of the brine migration test plan to include a firm definition of the experiment, an update of hardware development, and supporting calculations.

Recognizing the lack of liquid water in the Asse mine, the number of individual test sites to be tested at Asse was set at four, with two tests being conducted in each of the two major types of salt of interest. The other major variables are the presence or absence of a radioactive source and the gas pressure in the water collection volume. The electrical heating will be adjusted to give a maximum salt temperature of 210 C and a maximum thermal gradient of 3 C/cm, which are slight increases beyond typical repository conditions to accelerate the test somewhat without being nonrepresentative.

The proposed test assembly is shown in Figure 1. The temperature profile produced by the test is shown in Figure 2. Brine migration calculations have been performed for the two-year test duration using the average water content of Asse salt of 0.05 weight percent. For the liquid inclusion model, it was assumed that the water was present as liquid inclusions to provide a comparison with vapor transport. Salt permeabilities of 1.0 and 0.1 μ darcy were assumed for the vapor transport calculations, Figure 3. The depth into the salt from which water is collected is 11 cm for liquid inclusion migration, 60 cm for vapor transport with 1.0 μ darcy permeability, and 40 cm for 0.1 μ darcy. It is of interest that the calculated water magnitudes for vapor transport are significantly greater than for liquid migration, which was not expected. This will receive greater attention in the next period.

Agreement was reached with the FRG on the basic interface dimensions, particularly the source canister which must be compatible with German handling equipment. The FRG agreed to supply the radioactive source and will initiate a study in the next report period to complete definition of the handling equipment. The FRG also supplied geological descriptions of the proposed test site.

Waste-Rock Interaction

The bulk of the effort during this report period has focused on resolving the generic questions (see previous section) concerning the key issues and technical problems to be addressed by the waste-rock interaction test program. It has been found that a clear-cut distinction between the roles of laboratory programs and the field test program in addressing these issues and technical questions is mandatory. Ultimately, a separate issue of demonstration testing versus testing for purely technical data must also be resolved.

A multiple approach is being used to resolve these impediments to continued development of the test plan. Included are evaluations of prior and current laboratory programs bearing on waste-rock interactions, qualitative evaluations, and reviews of current nuclide migration models, and careful reviews of the basic issues generally agreed to be pertinent to salt repositories. The synthesis of this recent effort into conclusions suitable for incorporation into the existing test plan has not been completed.

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FIGURE 1. SECTION VIEW OF TEST ASSEMBLY

FIGURE 2. SALT FIELD ISOTHERMS

FIGURE 3. BRINE MIGRATION CALCULATIONS, VAPOR TRANSPORT, AND LIQUID INCLUSION MIGRATION MODELS

Technical Assistance

In November, Westinghouse (with the assistance of Stearns-Roger, Sandia, and D'Apollonia) began to develop field test requirements for the Salt Test Facility (STF). A draft list of tests to be performed in the STF, supported by technical justification, was developed. It reflects a mutually agreed philosophy of being limited to those tests that strictly require a facility like STF for proper performance or provide valuable data with minimal additional cost to the facility. The test planning was brought along to the point of providing a preliminary set of facility requirements for use by Stearns-Roger in facility design. In parallel with the test planning, a technical bases study was undertaken, beginning with an up-to-date tabulation of all important salt repository-related technical issues and a summary of both current status and ongoing or firmly planned work.

Future Activities

Brine Migration

More test cores will be driven into the proposed test site and the final brine migration test plan will be issued. Also, several supporting tests will be initiated to confirm that the equipment will perform as desired.

Waste-Rock Interaction

Work in the next quarter will stress clarification of the waste-salt interaction test plan basic objectives and related issues, including integration of the field tests with laboratory testing. Conceptual design of the test hardware will resume, supported by thermomechanical analyses of test hole behavior. These analyses, coordinated with the brine migration task, are necessary to confirm the adequacy of the test concepts and hole design.

Technical Assistance

During January, an updated set of Salt Test Facility requirements will be provided to Stearns-Roger, and drafts of the Conceptual Test Design and Technical Bases Documents will be provided to ONWI. Incorporation of comments and final issue of the two documents are scheduled for February. Interface support activities will continue as needed.

WBS 1.4.2

Project: Cooperative Field Tests in Salt

Principal Investigator: Sandia National Laboratories (T. O. Hunter)

ONWI Project Manager: W. F. Ubbes

Objective

This task supports the collection of field-test data necessary for Repository Data Base Development for salt, specifically: (1) to provide management support for field testing activities, (2) to evaluate and improve existing techniques for in situ rock characterization, (3) to perform experiments in southeastern New Mexico that address issues identified in the repository development programs, and (4) to support the international cooperative program on nuclear waste management.

Progress Reported Previously

The Cooperative Field Tests (CFT) program was instituted to coordinate various support activities provided by Sandia to ONWI in the repository data base development area. Formalization of the program was initiated in September 1980, and is designed to respond to issues regarding repository design data collection requirements. The CFT is broadly defined within four major categories: management, experimental technique evaluation, test series in Southeastern New Mexico, and support for international programs. This report is the first formal quarterly report prepared in the CFT.

Activities During the Reporting Period**Experimental Technique Evaluation**

Objectives. The objectives of this task are (1) to continue evaluation of techniques and systems currently available within the Sandia experimental program; (2) to investigate these systems within the in situ environment; and (3) to collect data required for repository design.

Procedures. Procedures for the operation of the radar-scanning and thermal-probe systems have been defined and are in the implementation, review, and update process.

Results. Circuitry modifications of the radar-scanning equipment have been completed, bench testing is satisfactory, and the system is ready. Laboratory check-out is completed, and plans are in progress to conduct scanning operations to locate trapped gas pockets in support of that experiment in the mine environment. The Radar Scanning Operation/Trapped Gas Experiment is scheduled to begin.

All in-mine activities associated with the Thermal Probe Evaluation experiment conducted at Avery Island have been completed and the data forwarded to Sandia by RE/SPEC. Analysis is in progress to reduce the data and to define additional tests that may be necessary. Preliminary assessment of the data indicates good consistency and agreement with previously obtained

conductivity data by RE/SPEC. The Avery Island drill holes have resulted in the lowest thermal resistance to data at the interface between the probe and the rock salt. Plans are in progress to extend measurements at a different power setting. More details will be presented when this initial test series is completed.

Data collection for the Stress Gage Evaluation continues at the Avery Island heater experiment.

Test Series in Southeastern New Mexico

Objective. The objective is to evaluate the impact of issues associated with repository data base development.

Procedures. Procedures for mine access and personnel training requirements have been defined and reviewed with the mine operators.

Results. The radiant heater/moisture release experiment in the Mississippi Chemical Potash Mine shut down on 12 November 1980, since all the experiment objectives had been attained. The electrical failure reported earlier was due to the corrosion and separation of an electrical connector external to the heater. This connection was replaced and the heater operated normally. The heater was then removed and examined; it was found to be in good shape. The borehole showed intrusion of salt crystal formations. Photographs were made of the borehole, and the intrusions and samples were removed for analysis.

The moisture yield for 113 days of operation was 188 ± 8 grams. Of this amount, 75 ± 4 grams were released in an over-temperature spike lasting about 10 days in which well temperatures exceeded 250 C. The moisture release rate was 0.4 grams/day during the last week of operation at a wall temperature of 180 ± 10 C. Elevated release rates up to 11 grams/day were noted for short periods after the heater was turned on or off.

In contrast, in the heater experiment at the Avery Island salt mine, the moisture yield for 138 days of heater operation at a wall temperature of 200 C was 12 ± 2 grams. The moisture release rate was 0.02 grams/day during the last week of operation. Elevated release rates up to 3 grams/day were noted for short periods after the heater was turned on or off.

Plans for conducting a follow-up experiment are in progress. System redesign will include the use of boroscopes to monitor crystal growth during the heating phase. The test plan is in progress. Fielding of the experiment is planned for early April, 1981. Results from the previous experiment are being assembled for publication.

The test plan for the Trapped Gas Experiment has been drafted and is in review. These tests will be conducted in conjunction with the Radar Scanning Experiment to detect and localize trapped gas pockets.

Support for International Programs

Objective. The objective is to support international agreements and cooperation between the U.S. DOE and foreign agencies in the waste management field.

Results. A radionuclide migration field test is planned in the Asse Mine in West Germany. Westinghouse, contracted by ONWI to design this field test, will try to modify the current test descriptions so that they are more suitable for a mine environment as opposed to a laboratory or hot-cell test.

A visiting U.S. scientist at the Bundesanstalt für Geowissenschaften und Rohstoffe (BGR) in the FRG is participating in discussions and exchanges with BGR personnel concerning (1) ongoing salt testing and modeling; (2) standardized testing, material characterization, and exchangeable data banks as proposed in the international agreement with the FRG; (3) the collection of field data to evaluate ground motions predicted for intermediate and long periods; and (4) the planning of common experiments at the BGR to identify deformation mechanisms and phenomenological characteristics of salt creep at low temperature and stress, including development of reference test series.

Contact was made with Hahn Meitner Institut (HMI) to develop plans for data transmittal from Sandia to HMI. A Telex was forwarded to the Bundesministerium für Forschung und Technologie explaining the U.S. position. While a response from FRG on U.S. participation in Gorleben modeling is expected, Sandia is preparing the data package. HMI will not use the data until spring, 1981.

Sandia has forwarded to BGR copies of the results of the first benchmark calculations using thermomechanical codes. Response from the BGR indicates that they will participate in these calculations. They intend to complete calculations by 15 April, 1981. Sandia will continue to coordinate this activity with the FRG.

Laboratory distribution coefficients of the Asse backfill samples are about 90 percent complete, and data have been forwarded to the Institut für Tieflagerung (IFT).

Sandia has been contacted by members of the IFT to provide a metallic overpack for a corrosion test in the Asse Mine. Plans are being prepared for provision of a titanium overpack in late 1981.

Future Activities

During the next quarter it is planned to (1) continue CFT management development, and (2) prepare and initiate test planning for additional CFT activities, including backfill experiments, permeability testing, and sealing efforts.

WBS 1.4.2

Project: Rock Behavior Under Elevated Pressure and Temperature

Principal Investigator: Lawrence Livermore Laboratory (H. C. Heard)

ONWI Project Manager: M. M. Lemcoe

Objective

The overall objective of the program is to develop an understanding of rock behavior under elevated pressure and temperature such that the response in space and time of a mined cavity in these rocks can be predicted. In order to do this, measurements are made in the laboratory at simulated, deep in situ conditions. It is expected that these data will be correlated with field observations and later incorporated into a model for prediction.

For Thermal Properties of Rocks, the objective is to determine the thermal conductivity, thermal diffusivity, and thermal expansion of candidate waste repository rocks at in situ conditions of temperature, pressure, and fluid content.

For Water Transport: Permeability and Related Measurements, the objective is to make systematic measurements of the permeability of intact and fractured rock as a function of confining pressure, pore pressure, and deviatoric stress. In order to extend the usefulness of the data, simultaneous measurements of fracture displacement, strain, electrical conductivity, and acoustic velocity and amplitude will also be carried out.

For Pressure and Temperature Dependence of Rock Elastic Constants, the objective is to determine the elastic moduli of a suite of candidate repository rocks as a function of temperature and pressure. These measurements, including Young's modulus, bulk modulus, and thermal expansion, are necessary to understand and predict repository behavior after emplacement of the waste.

Progress Reported Previously**Thermal Properties of Rocks**

This task started in August 1977. Making measurements under the conditions of temperature (to 300 C) and pressure (to 200 MPa) in the presence of pore fluid required an advance in measurement state of the art for rocks; an apparatus was designed. In the course of debugging it, a run was made on an unjacketed aluminum sample that showed the basic design to be viable.

Salt was found to have little abnormal behavior associated with the presence of confining pressure. Thermal conductivity and diffusivity generally decreased with increasing temperature in the manner seen in previous testing at room pressure. Thermal expansion showed no dependence upon confining pressure.

Experience with the two very successful runs on salt have provided the groundwork for a number of minor design changes aimed at speeding up and facilitating the experimental assembly and disassembly and at improving data accuracy and precision. These changes were accomplished before the first run was begun on Climax Stock quartz monzonite (CSQM). Two additional Climax samples and a fused silica standard sample have been about one-half prepared.

Water Transport: Permeability and Related Measurements

This task started in June 1976. A summary of progress through June 1980 includes the following major items:

- Design, construction and placement into operation of two complete sets of apparatus to determine permeabilities of large rock samples (15 cm diameter by 28 cm long) in the range of 10^{-11} to 10^{-24} m². These data are taken simultaneously with the principal imposed stresses and strains, acoustic velocities and pulse amplitudes, electrical conductivities, and displacement of a fracture (if present). A microprocessor system controls the test and records the data.
- Measurements of the above type were completed on intact samples of Westerly granite, White Lake gneissic granite, and Creighton gabbro, and on fractured Westerly granite, Stripa granite, and Creighton gabbro.
- Failure envelope data were completed on unfractured Westerly granite, Stripa granite, White Lake gneissic granite, Montello granite, Climax quartz monzonite, and Creighton gabbro. One sample of Stripa granite and one of Montello granite have been prepared. One of the latter type of granite was fractured. The test on the fractured Climax sample was about 40 percent complete.
- The hot water and gas permeability apparatus was modified to improve experimental control, and 20 NX-size samples were prepared for use in it. The test specimens include samples of Westerly, Stripa, and Climax granite rocks.

Pressure-Temperature Dependence of Rock Elastic Constants

This task started in June 1979. A summary of progress through September 1980 includes the following major items:

- Determination of the thermal expansion characteristics of Climax quartz monzonite to pressures of 27.6 MPa at 300 C. (See the results in Heard, 1980.)
- Multiple test samples of Stripa and Westerly granites, Creighton gabbro, and CSQM were prepared for testing to 55 MPa and 350 C. This involved three samples of each rock cored in each of three orthogonal directions.
- All measurements were completed on three samples each in the three orthogonal directions of the Stripa and Westerly granites.

The results are summarized as follows:

The coefficient of thermal linear expansion (α), Young's modulus (E), and bulk modulus (K) were determined for the Westerly and Stripa granites to temperatures (T) of 350 C and pressures (P) to 55 MPa. Using conventional triaxial apparatus, displacement measurements were made on three samples from each of three orthogonal directions for both rocks. Comparison between the directional values at any P, T, and nine from the sample population indicated that both granites are isotropic in α , E, and K. Values for α were neither constant nor a monotonic function of P or T. In both granites over the P range investigated, α typically increased from 6 to $12 \cdot 10^{-6} \text{ C}^{-1}$ at 40 C to 10 to $16 \cdot 10^{-6} \text{ C}^{-1}$ at 325 C. In both rocks over the T range investigated, increasing P from 6 to 55 MPa generally decreased α by 1 to $5 \cdot 10^{-6} \text{ C}^{-1}$.

Both E and K for both rocks decreased with T and increased with P in a nonlinear fashion. From 19 to 350 C, E decreased by up to a factor of 2, and K decreased by 2 to 3, depending on P. From 6 to 55 MPa, E increased by 3 to 6 and K increased by 3 to 5, depending on T. Assuming elasticity, the calculated Poisson's ratio ranged from 0.25 to 0.40 at 19 C, decreasing to 0.15 to 0.25 at 350 C (depending on both the rock and P). In both rocks, all measurements are consistent with microcracks controlling the thermoelastic response by opening with T and closing with P.

Activities During the Reporting Period

Thermal Properties of Rocks

A run was made during the quarter on a sample of Climax Stock quartz monzonite. The run was marked by exceptionally good behavior of the experimental system and, along with the thermal properties data gathered, allowed a careful analysis of the sensitivity of the data to slight changes in temperature conditions. Results of this run (which produced thermal conductivity, thermal diffusivity, and thermal expansion data to 200 C, 50 MPa) are illustrated in Figure 1. This run was terminated prematurely when the copper jacket reacted with pyrite inclusions in the rock and caused the jacket to leak. Another sample of CSQM, with stainless steel lining the inner and outer copper jackets, is in the final stages of preparation.

The thermal conductivity of Avery Island Salt was found to be independent of pressure from 0 to 50 MPa. The anomalous rise of thermal diffusivity at 10 MPa remains as previously reported.

Water Transport: Permeability and Related Measurements

During the quarter, the following subtasks have been worked on: (1) completion of the permeability test on fractured CSQM and reduction of all data (the equivalent permeability is 10^{-14} to 10^{-15} m^2 (10^{-2} to 10^{-3} D) at effective pressures of 2 to 25 MPa—qualitatively similar to the Westerly and Stripa granites); (2) testing of the intact CSQM sample; (3) completion of all sample preparations on the intact Stripa granite sample; and (4) continuation of the permeability testing of 6-cm-diameter intact Climax and Stripa samples at 20 C in the multisample apparatus—preliminary results indicate permeabilities of both rocks near 10^{-20} m^2 (10^{-8} D).

FIGURE 1. THERMAL CONDUCTIVITY, CLIMAX STOCK QUARTZ MONZONITE

Pressure-Temperature Dependence of Rock Elastic Constants

During the quarter, all measurements necessary to determine thermal expansion, Young's modulus, and compressibility on CSQM in three orthogonal directions were completed, and the data have been reduced. It can now be demonstrated how each quantity changes with pressure to 55 MPa and with temperature to greater than 350 C. Similar testing on Sudbury gabbro (to 300 C only) is about two-thirds completed.

Results of this work are as follows. Both rocks are isotropic in E , K , and α . For each rock, both E and K decreased with T and increased with P in a nonlinear manner. For example, in the monzonite, E and K decreased by up to a factor of 5 as P decreased from 55 to 6 MPa isothermally or as T increased from 19 to 500 C isobarically. Preliminary data for the gabbro showed similar though less dramatic decreases in both moduli at T to 300 C and over the same P range.

The values of a α for both the monzonite and gabbro increased with T and decreased with P in a nonmonotonic fashion. Depending on P, α in the monzonite increased from 8 to 11·10⁻⁶ C⁻¹ at 40 C to 22 to 25·10⁻⁶ C⁻¹ at 480 C. Calculations of the permeability of the monzonite based on the determinations indicated that the permeability may increase by up to a factor of 9 over the T range investigated. In the gabbro, values for α and the resulting increase in permeability can be expected to be less dramatic. In both rocks, all measurements are consistent with microcracks controlling the thermoelastic response by opening with T and closing with stress and P. Such determinations of E, K, and α are necessary for input to computational models for prediction of the thermomechanical response of a high-level radioactive waste repository constructed at depth.

REFERENCE

Heard, H. C., 1980. "Thermal Expansion and Inferred Permeability of Climax Quartz Monzonite to 300 C and 27.6 MPa," *International Journal of Rock Mechanics and Mining Sciences*, Vol. 17, pp. 280-296.

WBS 1.4.2**Project:** Fractured Rock Studies Program**Principal Investigator:** Lawrence Berkeley Laboratory**ONWI Project Manager:** M. R. Wigley**Testing Facilities for Ultra-Large Rock Cores**
(D. Watkins)**Objective**

The principal objectives of the program are: to develop, operate, and maintain a facility for testing ultra-large specimens of rock; and currently, to investigate the hydraulic properties of fractures as they pertain to fluid transport through the rock mass surrounding underground openings. The facility consists of a large triaxial testing machine and associated control, data acquisition, and specimen preparation equipment. The testing machine can accommodate cylindrical specimens, on the order of 1 meter in diameter by 2 meters high, which can be subjected to axial loads up to 17.8 meganewtons and hydrostatic confining pressures of 5.2 MPa. The test facility provides a general capability for testing ultra-large rock specimens for a variety of purposes in support of the research and technology development needs of the Geologic Repository Program of ONWI. Current research is concerned with the study of the basic phenomenology controlling the flow of fluid through fractures in rock, with specific emphasis on (1) the relationship between fracture permeability and stress in the rock mass, and (2) the influence of specimen size on the measured hydraulic properties of fractured rock.

Progress Reported Previously

The Large-Core Test Facility was constructed in 1967 for testing of rock-fill materials. Onsite activity by LBL began in October 1978. To adapt and upgrade the facility for research on fractured rock, LBL designed and installed an electro-hydraulic servosystem on the large testing machine to provide axial load and displacement control. An advanced data acquisition system (DAS) was also installed to record and analyze the large volumes of data generated by the experimental programs.

In support of the Swedish-American Cooperative Program on Radioactive Waste Storage in mined caverns in crystalline rock, a testing program to study the strength, deformational, and hydraulic properties of a 1-meter diameter by 2-meter high specimen of fractured granitic rock was completed in March 1980. Current research is directed at studying the effects of specimen size on the measured hydraulic properties of discrete fractures in granite. The test program involves measuring the conductivity of a natural fracture in a 0.91-meter diameter by 1.83-meter high cylindrical sample of Charcoal Black granite. Tests are being conducted in the large testing machine by injecting water into and withdrawing it from a borehole through the axis of the specimen. Measurements of hydraulic properties obtained in this manner are repeated at each of several stages of axial loading and unloading to study the relationship between fracture conductivity, fracture deformation, and normal stress. After completion of each test series, the specimen will be reduced in size and the tests repeated to investigate the effects of specimen size on these parameters.

By the end of FY 80, preparation and instrumentation of the 0.91-meter diameter Charcoal Black specimen was complete, and the rock was set up in the triaxial machine ready for testing. Figure 1 shows the triaxial vessel being placed over the sample. Figure 2 maps the fractures present in this specimen and shows the geometry of fracture B, which is the test subject.

**FIGURE 1. LARGE SPECIMEN OF CHARCOAL BLACK GRANITE
SETUP IN TESTING MACHINE**

Activities During the Reporting Period

Objectives

Program objectives for the first quarter of FY 81 were to: (1) complete testing of the Charcoal Black granite specimen at the 0.91-meter diameter size, (2) initiate analysis of the results from the first test series, (3) reduce the specimen size to 0.76 meter in diameter by 1.52 meters high, and (4) test the newly installed advanced DAS.

Procedures

The hydraulic properties of the natural fracture in the granitic specimen were investigated by a series of injection and withdrawal permeability tests conducted at each of several stages of applied axial stress during a total of five loading and unloading cycles in the large triaxial

FIGURE 2. FRACTURE MAP FROM CHARCOAL BLACK GRANITE

testing machine. Water pressure in the triaxial vessel was maintained close to 2.07 MPa throughout the experiment to ensure that any air entrained in the test water would remain in solution. Fracture deformations and rock strains were monitored through a system of instrumentation consisting of (1) linear variable differential transformers (LVDTs) mounted across the fracture plane and anchored near the top and bottom of the specimen, and (2) strain gages cemented to the intact rock. Preliminary reduction and analysis of the data obtained were completed, and detailed evaluations are currently in progress.

On completion of the test series, the specimen was removed from the test facility and reduced to 0.76 meter in diameter by 1.52 meters high in size by wire saw cutting on a large table jig operated by the Cold Spring Granite Company.

Results

The principal results obtained from the test program are summarized in Figure 3, showing the relationship between unit flow rate (the rate of flow of water through the fracture per unit of applied injection pressure) and applied axial stress. These results also illustrate the cumulative effects of cyclical loading and unloading. The normal stress path followed during the test is indicated by the arrows. Reduction of fracture conductivity due to loading is not fully recovered in unloading, and conductivity continues to decrease with each successive loading cycle.

The trials of the advanced DAS successfully demonstrated system hardware and software and will permit completion of the system to fully operational status for the next test series.

Conclusions

Analysis of the test results show the hydraulic properties of the natural fracture in the granite specimen to be sensitive to the magnitude and history of applied normal stress. Initial unit flow rates of 54×10^{-4} cm³/sec/cm-head of water decreased rapidly with increasing normal stress and after several loading cycles reached a minimum of $0.41 \text{ cm} \times 10^{-4}$ cm³/sec/cm-head at the maximum applied stress of 21 MPa. This represents an approximately 110-fold decrease in conductivity. A permanent decrease in conductivity occurs with each cycle of applied load; these effects are cumulative. For example, the unit flow rate measured at a normal stress of 3.5 MPa decreased from 2.07×10^{-4} cm³/sec/cm-head to 0.46×10^{-4} cm³/sec/cm-head when measured at the same stress level after five loading and unloading cycles. These effects will tend to mask any changes in fracture conductivity resulting from reduction in specimen size; further detailed study of this problem is required.

Problems

During conductivity testing of the Charcoal Black granite specimen at low applied stress, when the differential head applied to force water through the fracture must be kept low, flow rates were close to the lower limit of the operating range of the flow meters. If flow rates are significantly reduced during the next test series, it will not be possible to reliably measure flows in the low stress range with the equipment presently available. Alternative methods of flow measurement are being considered and test programs are being re-evaluated to assess

FIGURE 3. RESULTS OF DIVERGENT HYDRAULIC FLOW TESTS ON A NATURAL FRACTURE IN A 0.914-METER DIAMETER SAMPLE OF CHARCOAL BLACK GRANITE

the importance of fracture permeability tests conducted in the very low stress range. Current plans are to proceed with the next test series with the present equipment so that the range and resolution of new equipment required, if any, can be determined.

The time required for a given flow test to stabilize was unexpectedly found to be on the order of two hours per test. This was due to an exponential flow transient of long duration that occurred before steady-state conditions were reached. Post-test analysis indicates that major portions of this transient can be attributed to stress relaxation in the hydraulic hoses used in the water injection system. As far as is practical, this system will be hard plumbed for the next test series. It is expected that test durations will then be reduced and permit more flow tests to be performed during each loading and unloading cycle with consequent improvement in data resolution.

Future Activities

Principal program objectives for the next quarter are:

1. Completion of interface hardware and improved instrumentation to bring the advanced DAS to full operation,
2. Preparation, instrumentation, and testing of the Charcoal Black specimen at reduced 0.76-meter diameter size, and,

3. Continued analysis of data from the first test series on the 0.9-meter diameter specimen size and comparison of these results with data obtained from the 0.76-meter diameter specimen size.

Large Triaxial Cell Site Usage and Safety Investigation (A. DuBois)

Objective

The objective of this task is the resolution of the crucial site usage and safety questions connected with the construction of a new high-pressure/high-temperature triaxial testing cell for ultra-large rock cores at the Richmond Field Station.

Activities During the Reporting Period

No progress report was submitted for the period.

WBS 1.4.2**Project:** Materials Properties and Stress Measurement Techniques**Principal Investigator:** Lawrence Berkeley Laboratory**ONWI Project Manager:** M. M. Lemcoe**Materials Properties Testing Program****Task A: Small-Core Test Facility**

(N. Cook and L. Myer)

Objective

In situ experiments in the Stripa Mine, Sweden, have yielded a great volume of data on the thermomechanical response of a granite rock mass to the heat that might be produced by buried nuclear waste. For proper analysis of the data from these experiments, it is necessary to have additional data in the form of thermomechanical properties obtained from rock samples tested in the laboratory. It is the objective of the materials properties test program to study and acquire thermomechanical properties (e.g., Young's modulus, Poisson's ratio, and thermal coefficient of expansion) using drill core from the Stripa Mine experimental areas.

Progress Reported Previously

The materials testing program began in FY 79. Early in the program, work began on a stiff triaxial test machine capable of providing a confining pressure of 55 MPa, test temperatures up to 200 C, and a maximum axial load of 1.4 MN. The test cell will accommodate either 56 or 62-mm-diameter core of a length-to-diameter ratio of 3 to 1.

Fabrication and installation of the test apparatus in the laboratory was completed in the beginning of the second quarter of FY 80. This was followed by a period of transducer calibration and general machine shakedown which continued into the third quarter. Concurrently, a computer system for data acquisition and test machine control was installed. During the third quarter, suitable sample jacketing and strain-gaging techniques were also developed. Testing of intact samples of 62-mm-diameter core was begun in the fourth quarter of FY 80.

Activities During the Reporting Period**Objective**

Testing of intact examples of 62-mm-diameter core continued. The objective was continued data collection for investigation of the effect of pressure and temperature on E (Young's modulus), ν (Poisson's ratio), and α_v (volumetric coefficient of thermal expansion) for intact, dry Stripa granite.

Procedures

To be representative of in situ conditions during the heater tests, the range of confining pressure was 2 to 55 MPa, while the temperature ranged from room temperature to 200 C. Each specimen was subjected to the entire pressure and temperature range with data being taken at many pressure-temperature (P-T) states. To minimize the permanent damage sustained by the rock at any P-T state, property measurements were begun at the highest confining pressure and lowest temperature. At each confining pressure the temperature was increased in 20 C increments, and thermal expansion measurements were taken at each increment after thermal equilibrium was attained. Stress-strain measurements for determination of E and ν were made after every second temperature increment. Sample deformations were monitored by strain gages applied directly to the specimen.

Initially it was necessary to interrupt the test sequence at the end of each working day by depressurizing and cooling the sample. Thus, it was not possible to achieve all P-T states in the sequence described in the test plan. This problem was eliminated in the second half of the quarter by implementing 24-hour operator coverage for the test machine.

Results

Discussion of results is limited to information obtained from one sample (No. EO30.24-0.45) because there has not been sufficient time to fully reduce data from the other tests. These results should, of course, be viewed as preliminary.

Stress-strain tests were performed at numerous P-T states to determine if the elastic moduli were dependent upon stress state or temperature. Increasing pressure resulted in two effects on this sample: (1) a decrease in the nonlinearity of the stress-strain curve and (2) an increase in the values of E and ν . For any given temperature state, the greatest nonlinearity occurred at the lowest confining pressure of 2 MPa. For example, at 25 C, and 2 MPa confining pressure, E varied from 60.5 GPa at a stress difference of $\sigma_1 - \sigma_3 = 20$ MPa to 67.4 GPa at a stress difference of $\sigma_1 - \sigma_3 = 80$ MPa. Above confining pressures of 15 MPa, both E and ν were constant during loading. Some hysteresis, leading to some nonlinearity, was noted on unloading at all confining pressures. The increase in moduli with confining pressure is also exemplified by the data at 25 C. Young's modulus increased from 60.5 GPa, as noted above, to 71.6 GPa under 55 MPa confinement. Similarly, ν increased from 0.15 to 0.22.

Effects of temperature changes on the moduli are illustrated in Figure 1. In general, as temperature was increased, values of E and ν decreased. However, as seen in Figure 1, almost no temperature effect was noted at 55 MPa confinement. Under 2 MPa confinement, the reduction in E was about 15 percent, whereas the reduction in ν was about 35 percent.

Strain gages permitted direct measurement of volumetric as well as linear (axial) thermal expansion of the samples. Figure 2 presents values of the volumetric coefficient of thermal expansion (α_v) as a function of temperature. (Values of α_v were determined by a polynomial regression analysis of the data.) As seen in Figure 2, α_v increases with increasing temperature. Under 2 MPa confinement, the increase was about 65 percent, whereas under 55 MPa confinement the increase was limited to about 40 percent.

FIGURE 1. TEMPERATURE DEPENDENCE OF YOUNG'S MODULUS AT 55 MPa AND 2 MPa CONFINING PRESSURE

For any given temperature, an increase in confining pressure resulted in general in a decrease in α_v . At 55 MPa confinement, α_v was about 10 percent to 15 percent less than under 2 MPa confinement.

After completion of the full test sequence, the sample was repressurized to a confining pressure of 30 MPa and the test sequence at that confining pressure was repeated. This test was done to determine if the exposure of the sample to the various P-T states and axial loadings caused sample damage which influenced measured properties. No significant difference was noted in values of E and ν . Values of α_v were less than 5 percent higher in the repeat test.

Conclusions

Results from one test indicate some dependence of E and ν on both temperature and stress state. Nonlinearity in the stress-strain curves and the temperature dependence of E and ν decreased as the confining pressure increased. The effect of confining pressure on E and ν was greatest at high temperature.

Dependence of α_v on temperature and pressure was also noted, although the temperature dependence was much more pronounced.

As not all the data have been thoroughly analyzed and some testing remains to be done, the results and conclusions reported herein should be considered preliminary.

FIGURE 2. TEMPERATURE DEPENDENCE OF VOLUMETRIC COEFFICIENT OF THERMAL EXPANSION AT 55 MPa AND 2 MPa CONFINING PRESSURE

Future Activities

Reduction of data from tests completed so far will continue and should be completed by mid-February. Concurrently, an electric servo system is being installed to allow computer control of test conditions. Installation of hardware will be completed before mid-February, and necessary software should be completed by the end of February. Upon installation of hardware, necessary machine recalibrations will be carried out. Testing should resume in the early part of March. At least two more intact samples of 62-mm-diameter core will be tested, followed by testing of samples containing healed fractures.

Materials Properties Testing Program
Task B: Stripa Ultra-Large Core Study
 (D. Watkins)

Objective

The permeability of the core will be investigated under varying states of axial stress. This will be achieved by injecting water into a borehole drilled through the long axis of the core while the core is subjected to axial loading in the ultra-large triaxial testing machine. Several

cycles of increasing-decreasing axial stress will be applied to study the effects of load history on permeability and deformation modulus. The tests will be designed to study the overall permeability of the core and the conductivity of selected principal fissures. The core will be instrumented to gain insight into the stress-deformation behavior of the principal fracture and its relationship to fracture conductivity. Measurement of total axial strain will yield information regarding the deformation modulus of fissured Stripa granite. Data of this type, obtained from a sample of dimensions closer to those of practical significance than have previously been available, are important to the rational interpretation of the in situ hydrological and geotechnical results gathered from the Stripa experiments.

Activities During the Reporting Period

No progress report was submitted for the period.

Stripa In Situ Stress Measurements (T. Doe)

Objective

The objectives of this program are twofold: (1) to provide in situ stress data for interpretation of the heater and hydrology experiments at Stripa and (2) to compare and assess two different methods of in situ stress measurement. During 1980 the goal was to perform both hydraulic fracturing and overcoring stress measurements in SBH-4, a vertical hole drilled from the surface north of the Stripa Mine. Both types of measurements were performed from depths 100 meters or less to 300 meters.

The data obtained from the SBH-4 work will provide far-field stress values for the thermomechanical modeling of the Stripa heater experiments. The objectives of the 1981 field work are to complement the SBH-4 far-field stress measurements with near-field measurements close to the underground test areas. Hydraulic fracturing will be used in addition to USBM gage, Leeman triaxial cell, and CSIRO gage methods as a means of combining the results of different measurement methods.

The near-field underground stress measurements will play a key role in validating the mechanical numerical model of the Stripa tests.

Progress Reported Previously

The project started in the third quarter of FY 80, when the field equipment was assembled. During the fourth quarter of FY 80, the hole SBH-4 was drilled and the stress measurements were performed.

Activities During the Reporting Period

Objective

The objective of the first quarter FY 81 work was to perform the laboratory tensile strength testing of the core from SBH-4 so that final analysis of the hydraulic fracturing stress measurements could be made.

Procedures

Hydraulic fracture tensile strength tests were performed on 45 specimens of SBH-4 core. The test procedure consisted of drilling a small borehole into a sample of SBH-4 core 2.5 inches in diameter and 5 inches long. The holes were drilled to within 1 inch of the end of the specimen, and a length of stainless steel tubing was cemented into the specimen to a depth of 1 inch. The stainless tubing was inserted into a special end cap for conducting fluid to the small borehole. Each specimen was stressed axially in a loading frame to a stress equal to that of the overburden at the depth from which the sample was taken. Water was then injected into the rock at constant flow rate with an electrically driven pressure intensifier until the sample ruptured. A Dunegan-Endevis acoustic emission monitoring system was used to monitor acoustic activity during injection to determine whether or not crack growth was initiated prior to the loss of fluid pressure in the sample.

Two different borehole sizes were tested, 1/4 inch and 1/2 inch, to determine if size effects exist.

Results

Three different textural rock types were identifiable in the core: granite without healed fractures, granite with epidote or calcite-filled healed fractures, and granite with quartz or chlorite healed fractures. The data from the tests grouped in two modes, one for the intact and quartz or chlorite healed fractures, the other for the epidote healed fractures. The reduction in strength for the epidote fractures was about 30 percent for the 1/4-inch hole specimens and 16 percent for the 1/2-inch specimens. Overall, the strengths for the 1/2 inch specimens are 30 percent lower than the comparable 1/4-inch hole specimens.

Conclusions

The tensile strength results have not yet been incorporated into the analysis of the SBH-4 stress measurements. A value of 1,500 psi was assumed as the tensile strength value in the previously reported preliminary analysis; once the size effect for 3-inch holes is determined, the actual tensile strength will be slightly lower.

Healed fractures, which were unavoidable in selection of test zones in the SBH-4 hole, appear to have only a minor effect in reducing tensile strength.

Future Activities

The tensile strengths will be incorporated into the SBH-4 stress results and a report completed about April 1. Planning activities for the underground measurements, including work by the Colorado School of Mines and the University of Luleå (Sweden) will be completed for the July 1981 work.

Post Heater Drill-Back Investigation
(H. Wollenberg)**Objective**

The objective of this task is to investigate the effects of the now complete heating/cooling cycle on the Stripa H-10 heater borehole and adjacent granite rock mass. This is to be accomplished by the drilling and TV logging of two new boreholes intersecting the heater location. Core samples will be shipped to Berkeley for laboratory examinations.

During FY 81, the project will be completed by continuing the petrographic study of the post-test cores to detect changes in fracture parameters, microfracture intensity, fracture-filling material, and rock-forming minerals. Mechanical properties such as Young's modulus, Poisson's ratio, and the coefficient of thermal expansion for fractured and intact core samples will be measured for comparison with pretest samples. The heater hole photographic data will be analyzed to relate the three-dimensional spalling pattern to the theoretical thermal stress.

Progress Reported Previously

During the latter half of FY 80, this project obtained new core samples traversing the heated and subsequently cooled rock in the heater test area, and initiated a petrographic examination using thin sections. In addition, stereoscopic photographs were made of the interior walls of heater holes H-9 and H-10 to study decrepitation and spalling.

Activities During the Reporting Period

No activity took place nor is any planned for this project until April 1981.

WBS 1.4.2

Project: Swedish Cooperative Project

Principal Investigator: Lawrence Berkeley Laboratory (N. Cook)

ONWI Project Manager: W. F. Ubbes

Objective

The project objective is to assure long-term containment of potential contaminants during the underground storage of nuclear waste in crystalline rock. Whether these conditions are met depends on the coupled interactions of heat flow, fluid flow, and rock mass properties. Lawrence Berkeley Laboratory, in conjunction with the Swedish Nuclear Fuel Safety Board, is conducting in situ heater tests, hydrologic tests, and fracture system characterization studies in granite in an iron mine in Stripa, Sweden. Additional support activities include:

1. Defining the fracture network through the use of borehole camera logging and geophysical testing
2. Measuring hydrological and material properties of the Stripa granite in the laboratory
3. Macroporosity studies underground in the ventilation drift
4. Continued development, improvement, and verification of existing mathematical models
5. In situ stress measurements through the overcoring and hydrofracturing techniques.

Progress Reported Previously

Previously reported progress includes the recalibration of the various instruments that were used during the heater experiments, preparation of a master data file for these experiments, progress in the fracture hydrology program, and up-to-date results from the ongoing macroporosity investigation at Stripa.

Activities During the Reporting Period**Instrumentation**

The thermocouple calibrations, done at Stripa before and after the heater experiments, were plotted and examined. The plots indicate that typical scatter in temperature calibrations, relative to NBS calibrations for type K thermocouples, falls within +1 C up to 100 C and within +2 C up to 200 C. Since the NBS curve was used to process the temperature data, this approach will set the range of measurement error for observed temperatures in the heater experiments.

Compilation of data on a hole-by-hole basis commenced during the quarter and is about 50 percent complete at this time. IRAD data and extensometer data are being compiled first. The data, compiled in a set of notebooks, will furnish a ready reference for calibrations, instrument down-time, instrument maintenance, and quirks in data that can be traced to operational causes. In addition, the postexperiment extensometer calibrations, which are quite detailed, were compiled and analyzed. Repeatability with pre-experiment calibrations was in general quite good.

One final field check on survey results was undertaken and completed during the quarter when discrepancies were noted between existing measurements on borehole collar elevations and consequently, on some instrument locations. The discrepancies are on the order of 1 to 2 cm.

A report on the results from the nine different geophysical borehole probes run in borehole SBH-1 was completed during the quarter. It describes the application of borehole measurements for rock typing, mineral identification, porosity estimation, and fracture detection. The effect of a relatively minor mafic mineral (chlorite) is quite dramatic on the neutron and density logs, and must be accounted for before porosity can be estimated. Porosity increases of a few percent are not uncommon around fracture zones, based on a cross-plot interpretation of resistivity and neutron data.

Work progressed on the ultrasonic data base. Limitations and some low-level effects introduced by the direct record process on the instrumentation tape recorder were identified and assessed for their importance in future spectral processing.

Future Activities

A report on thermocouple, extensometer, and heater performance will be drafted during the next quarter.

WBS 1.4.2

Project: Stripa Testing Project

Principal Investigator: Swedish Nuclear Fuel Supply Co., Safety Board Division (SKBF/KBS)
(Lars Nilsson, Coordinator)

ONWI Project Manager: R. A. Robinson

Hydrogeological Investigations in Boreholes

(L. Carlsson and T. Olsson, Geological Survey of Sweden)

Objective

The objectives of the hydrogeological investigations in boreholes are as follows:

1. Methodology development for hydrogeological and hydrogeochemical investigations in subsurface, nearly horizontal, and vertical boreholes.
2. Instrumentation and equipment development in subsurface, nearly horizontal, and vertical boreholes
3. Hydraulic, chemical, and isotopic characterization of the Stripa granite and ground waters.

Activities During the Reporting Period

During the reported time period, the drilling down to the first break at 150 meters was completed. The drilling was carried out without any tracer added to the drilling water. The rock down to the first stop is exclusively made up of a homogeneous granite with only few fractures. Mean fracture frequency is about 2.4 fractures per meter, but most of the fractures are sealed by chlorite, and calcite and quartz also occur. The frequency of opened fractures was about 1.5 fractures per meter. From the core logging it seems probable that water inflow to the borehole so far takes place only at four levels within the 150 meters.

After the drilling was finished, the water flow from the borehole was found to be about 6 ml/min (0.006 liter/min), of which the major quantity probably originates from the 51.5 to 53.5, 55.5 to 57.5, and 66 to 71.5-meter levels. It is estimated that the flow is too low to allow any more extensive water sampling for hydrogeochemical purposes.

Rock stress measurements were performed at the 150-meter level by the Swedish State Power Board, but so far no results have been presented. No accurate method exists for core orientation in hard rocks at depths greater than 100 meters.

It was decided that the sampling procedure for the vertical borehole would be divided into three different phases: sampling during drilling, sampling under pressurized conditions, and sampling under free-flowing conditions.

The members of the group of hydrogeochemists were asked to leave their comments and the reason for the chosen analyses, the quality and detection limits required, and how the hydrogeochemistry will aid the subject of nuclear waste disposal.

Future Activities

The following activities are planned: (1) hydraulic testing of the rock mass down to 150 meters; (2) sampling and analysis of water from the uppermost 150 meters (due to the very low water flow, the sampling will be made for a reduced number of constituents); (3) continued drilling in 150-meter intervals; and (4) rock stress measurements, hydraulic testing, and water sampling during each drilling break.

Migration in Single Fissure, Preparatory Study

(I. Neretnieks, Royal Institute of Technology, Sweden)

Objective

In this preparatory investigation various tracers, including particles of different sizes, are introduced into a single fissure in granite. The objective is to develop a reliable method for this type of investigation and to test the injection and sampling equipment that will be used in the main investigation not yet funded by the Stripa Testing Project. The proposed migration distance is 5 meters.

Activities During the Reporting Period

The water-bearing fissure was located, and circulation of fluorescein was initiated. The test was speeded up by increasing the injection pressure above the natural pressure in the fissure. Thereafter, the pressure in the injection hole was allowed to decrease to stability. After cleaning the hole with anoxid water, Rhodamine Wt was circulated for 7 days. Anoxid water was again used for cleaning the hole, and preparations for injection of fluorescein and Rhodamine Wt simultaneously started.

The water-bearing fissure is probably located 1.97 meters from the bottom of the injection hole. The time needed for fluorescein to reach the sampling holes was less than 24 hours, and water from three of the sampling holes contained tracers with a dilution factor varying from 25 to 400. Rhodamine Wt, that was circulated for 7 days, was not found in the sampling holes.

Future Activities

Both fluorescein and Rhodamine Wt will be injected simultaneously. A retardation of Rhodamine Wt during this test would contradict the results obtained from the laboratory sorption tests made with tracers and crushed rock. The preparatory study will be completed during February, 1981.

Buffer Mass Test
(R. Pusch, University of Luleå, Sweden)

Objective

The general objective of the buffer mass test (BMT) is to verify the suitability of the suggested buffer materials under real conditions on site.

Activities During the Reporting Period

The first part of the period consisted mainly of market surveys for optimum choice of temperature and moisture gages, while pilot borings and other field activities dominated the later part.

Field Work

Twelve pilot inspection holes were bored to obtain a structural (hydrological) mapping of the tunnel floor for the choice of heater hole sites. Water inflow measurements as well as core mapping suggest that the heater holes be located according to the original plans. One of the holes (No. 6) will be located in virtually dry rock, and this will give a good opportunity to observe water migration and possible drying effects in very dense rock. The rest of the holes will probably have considerable access to flowing ground water. The first 78-cm-diameter heater hole (No. 6) was also bored in mid-December to approximately 2 meters deep without considerable difficulties.

Other Activities

The preparation of preliminary prognoses with reference to the expected development of temperature fields and rate of moisture uptake was started in November. This activity will increase considerably when the results from the heater hole borings (water inflow measurements and core mapping) become available. Also, the results from previous laboratory tests on backfill materials are required as a basis, and these results are presently being analyzed.

Future Activities

The work for the next quarter will include the following: (1) drilling of the deposition holes will continue with increased efforts; (2) drilling of holes for thermocouples and water pressure cells; (3) geological and structural documentation of the storage holes; (4) predictions of temperatures, water uptake, and swelling pressure; and (5) compaction of bentonite and preparation of bentonite blocks for the heater holes.

WBS 1.4.2**Project:** Stripa Data Analysis**Principal Investigator:** Lawrence Berkeley Laboratory (T. Chan)**ONWI Project Manager:** M. R. Wigley**Objective**

The purpose of the analysis is to gain a better understanding of the response of a fractured granite rock mass to thermomechanical loading. This is to be accomplished by careful comparison of modeled temperature, displacement, and stress fields with measured data from the Stripa heater tests.

The program will attempt to assess the influence of the following factors on the Stripa field data collected during the heater tests conducted at Stripa, Sweden, from June, 1978 to June, 1980: (1) in situ state of stress; (2) presence of excavations; (3) the three-dimensional geometry of the experimental arrangement; (4) temperature, stress and, possibly, loading path dependence of rock properties; (5) presence of fractures; and (6) presence of ground water.

Progress Reported Previously

In preparation for post-test data analysis, the enormous data base (approximately 50 million measurements) has recently been verified, converted to engineering units, logically ordered, and written onto two standard magnetic tapes.

Activities During the Reporting Period**Objectives**

Program objectives for the first quarter of FY 81 included examination of temperature data to determine which thermocouples provide reliable data to be compared with numerical models; estimation of the effect of fractures on thermal conductivity of the rock mass; and analysis of the method of data interpretation for USBM borehole deformation gages.

Procedures

Thermal analysis of data from the full-scale experiment was initiated by separating unreliable measured data from reliable data. Shortly after heating started in June, 1978, many of the stainless steel clad thermocouples at lower locations produced unreliable data due to corrosion problems. The analysis shows that during this early period in the experiment, approximately 16 percent of the H-9 area data and approximately 47 percent of the H-10 area data were affected by corrosion. Due to the axial symmetry of the experiment and the otherwise good data, this loss of early-time temperature data should not pose a serious threat to the overall thermal analysis. Other possible sources of measurement error are being examined.

The bulk thermal conductivity of a fractured rock mass was estimated using theoretical correlations of thermal conductivity of granular materials. Two types of models were used: (1) "Flux law models", based on direct solution of the heat conduction equation, and (2) "Ohm's law models", based on electrical analogs. For a rock mass with 2 percent porosity (due to voids, cracks, and fractures), the flux law model predicts that if all the pore space is filled with bulk thermal conductivity, k_1 will be 3 percent lower than k_s . Ohm's law models predict k_1 to be 5 percent lower than k_s .

Diametral displacements of the USBM gage boreholes corresponding to numerically modeled stresses were calculated by both ignoring and including the strain in the axial direction of the borehole. In the Stripa heater tests, the axial strain from thermal loading significantly affects the borehole diametral displacements, especially for horizontal boreholes. This is illustrated in Figure 1, where the curve labeled "triaxial" represents the calculated diametral displacement with axial strains properly taken into account. Clearly, the "triaxial" curve differs radically from the "biaxial" curve obtained by assuming plane strain condition. In this case, the "triaxial" curve also agrees better with experimental data. It should be noted that, regardless of what analytical or numerical model is used for the calculations, the effect of the axial strain cannot be neglected.

FIGURE 1. TIME HISTORY OF DIAMETRAL DISPLACEMENT OF A 38 mm BOREHOLE AT RADIAL DISTANCE 2.99 m FROM THE 5 kW HEATER, 0.62 mm ABOVE MIDPLANE

Displacements calculated by biaxial and triaxial theories are shown, along with field data measured by USBM borehole deformation gage U 25 (Component 3). Positive sign indicates reduction of borehole diameter.

Conclusions

From the results of the bulk thermal conductivity estimates and the diametral displacement calculations, it is concluded that (1) fractures have little effect on the bulk thermal conductivity of the granite rock mass, and (2) the three-dimensional nature of thermal stresses must be considered when interpreting borehole deformation data.

Future Activities

During the next quarter, the following activities are planned:

1. Continue thermal analysis.
2. Perform analysis to assess the possible influence of mining excavations on the far field in situ stress measurements.
3. Initiate work to calculate the in situ and excavation-induced stresses in the area of the heater tests.
4. Initiate work to assess the sensitivity of the in situ thermomechanical response to (a) rock properties determined at various laboratories and (b) geometrical distribution of "open" fractures.

WBS 1.4.2

Project: Expected Repository Environments in Granite (ERE-G)

Principal Investigator: RE/SPEC, Inc. (J. D. Osnes)

ONWI Project Manager: G. E. Raines

Objective

The objective of this study is to establish the expected repository environments in a generic granitic repository containing commercial high-level waste, defense high-level waste, or spent fuel. The expected repository environments will be described in terms of transient temperature fields, deformations, resultant stress states and regions of strength failure, and perturbations to ground-water flow. In addition to these states or environments, consideration will also be given to describing chemical and nuclear radiation environments and vapor pressures in the waste package drillholes and in the disposal rooms.

This work supports and addresses particular problems relative to the Rock Mechanics Program of the NWTS program. In particular, the ERE-G study provides technical bases for reports authored by the Reference Repository Conditions Interface Working Group (RRC-IWG), an ad hoc IWG established by the Isolation Interface Control Board.

Progress Reported Previously

The ERE-G study was initiated under the auspices of ONWI on 1 October 1979. During FY 80, thermal analyses of the canister near-field, room-and-pillar, and far-field regions were performed, and nuclear radiation environments and vapor pressures were calculated. Reference chemical and mineralogical compositions of granite and reference ground-water compositions were also established. A report describing the results of these activities was prepared and is currently in draft.

Activities During the Reporting Period

All ERE-G activity during this quarter was related directly to participation in the RRC-IWG, and no further technical activity was undertaken. Activities in support of the RRC-IWG included submission of *Interim Reference Repository Conditions for Commercial and Defense High-Level Nuclear Waste and Spent Fuel Repositories in Granite* to the RRC-IWG, which approved this draft and submitted it to ONWI management review. Drafts of similar reports for salt, shale, and tuff were also reviewed as a part of ERE-G activities.

Future Activities

In mid-February, thermomechanical modeling of the reference repositories in granite will begin. Also, an attempt to further define geochemical environments will be made, particularly with respect to temperature, pressure, and nuclear radiation conditions expected in the reference repositories. These activities will continue through the next two quarters.

WBS 1.4.2**Project:** Heated Block Test in Granitic Gneiss**Principal Investigator:** Terra Tek, Inc. (M. D. Voegelé)**ONWI Project Manager:** W. F. Ubbes**Objective**

The project objective is to determine the in situ mechanical, thermal, and hydrologic properties of a jointed crystalline rock under conditions of varying temperature and stress.

Progress Reported Previously

The current project began in July, 1979. Extensive preparatory work included the design, specification, and laboratory calibration of instruments to measure stress, strain, displacement, ultrasonic velocity, and joint permeability. Baseline in situ data were obtained before drilling the flatjack slots. Following installation of the 16 meters of flatjacks (in duplicate to allow for failures) around the cubic block of jointed gneiss, the main series of ambient loading tests was completed. At the time of writing, one heating cycle under biaxial stress is nearing completion following cooldown.

Results summarized here incorporate some of the ambient loading test results consisting of biaxial, uniaxial north-south, and uniaxial east-west loading cycles to 6.9 MPa. The stress and deformation dependence of the following parameters are summarized: static deformation moduli, dynamic moduli, and joint permeability.

Activities During the Reporting Period

The main activity has been the performance of three different load-path cycles under ambient (12 C) conditions. The biaxial loading puts the three sets of vertical joints intersecting the block in a state of compression, while the N-S uniaxial load cycle puts the instrumented diagonal joint set in a state of shear. The subsequent E-W uniaxial load cycle reverses the shear direction. The resultant effects of joint closure, opening, and shear are clearly reflected in the results.

Static Deformation Moduli

Whittemore gage pins distributed across the whole surface of the block indicated maximum east moduli in the range of 80 to 90, 100 to 125, and 25 to 50 GPa during the biaxial and the N-S and E-W uniaxial tests, respectively. Average intact rock moduli measured on core taken perpendicular and parallel to the foliation (CSM laboratory test results) ranged from 50 to 66 GPa, with individual values as high as 94 GPa. A very hard quartz lens crossing the S-W corner of the block probably explains the values above 100 GPa.

When the flatjacks are pressurized at opposite sides of the block, cracks occur in the grout above them due to shortening of the block (δ_b) and compression of the surrounding rockmass (δ_m). An elastic solution (Poulos and Davis, 1974) was used to estimate the overall deformation modulus of the block for comparison with the short base-length measurements:

$$E_{\text{block}} = \frac{\Delta P \cdot 1000}{\delta_b} = \frac{\Delta P \cdot 1920}{(\delta_m + \delta_b)},$$

where ΔP = change of pressure and $(\delta_m + \delta_b)$ = total change of crack width.

Analysis of the boundary cracks indicated mass deformation moduli of about 20 GPa on loading and values in the range of 30 to 60 GPa on unloading. The block was somewhat stiffer under biaxial load than under either uniaxial load paths.

Dynamic Moduli

Dynamic moduli obtained from crosshole ultrasonic tests ranged from 10 to 49 GPa at 50 cm deep, while at the base of the block (200 cm), values ranged from 27 to 68 GPa. The average dynamic moduli in the loaded directions were 53, 50, and 53 GPa, respectively; in the biaxial and N-S and E-W uniaxial tests at the highest stress level of 6.9 MPa. Significantly, the average dynamic moduli in the unloaded E-W and N-S directions in the uniaxial tests were 34 GPa and 47 GPa, respectively. Reversed shearing of the diagonal joints in the E-W test presumably tended to reduce their aperture nearly to biaxial levels.

Joint Permeability

The water injection tests performed in the weathered, mineralized, diagonal joint, while it was loaded and unloaded in the normal and shear directions, are of particular interest to repository evaluation. Water flow was monitored between a vertical injection hole and two parallel observation holes, each drilled down the plane of the joint. Figure 1 summarizes the results of the ambient injection tests. The numbers 1 through 8 indicate the order of testing. Theoretical smooth wall apertures (e) and permeabilities ($e^2/12$) can be calculated from the following well-known equation for flow between parallel plates:

$$q = \frac{de^3}{12\mu} \left(\frac{dP}{dy} \right)$$

Table 1 summarizes the most significant data.

An important feature of the results is the tendency for slightly increased joint permeability at high shear stress, despite the relatively high normal stress (3 MPa) and the modest shear displacement of 0.2 mm.

Future Activities

The first heating cycle to 50 and 80 C is now nearing completion. Preliminary results indicate reduced joint permeability at higher temperatures despite controlled boundary stresses. A further heating cycle at reduced biaxial pressure is scheduled.

FIGURE 1. EFFECT OF NORMAL STRESS ON FLOW RATE AND ON THE THEORETICAL SMOOTH WALL APERTURE (e)

1 Micron = 0.001 mm. These are ambient tests.

REFERENCE

Poulos, H. G. and E. H. Davis, 1974. *Elastic Solutions for Soil and Rock Mechanics*, John Wiley and Sons, Inc., New York, NY.

TABLE 1. STRESS DEPENDENCY OF JOINT PERMEABILITY

Test	Stresses on Joint		e, microns(a)	K, darcys(b)
	σ_n , MPa	τ , MPa		
Preslot Drilling	(in situ stress)		38	120
Postslot Drilling	0	0	61	310
Biaxial	0	0	56	261
	6.9	0	30	75
	0	0	46.8	183
N-S Uniaxial	0	0	46.8	183
	3.1	3.4	30	76
	0	0	47.4	187
E-W Uniaxial	0	0	47.5	187
	3.8	3.4	27	61
	0	0	53	234

(a) 1 micron = 0.001 mm.

(b) 1 darcy = 10^{-8} cm².

WBS 1.4.2

Project: Crystalline Rock Mining Technology

Principal Investigator: Colorado School of Mines (W. Hustrulid)

ONWI Project Manager: W. F. Ubbes

Objective

The Colorado School of Mines (CSM) has established a research facility at its Experimental Mine located at Idaho Springs, Colorado, to develop excavation techniques and site evaluation procedures appropriate for the location, design, construction, licensing, and long-term monitoring of a hard rock nuclear waste repository. In the experimental room, a large (2 x 2 x 2-meter) block isolated from the surrounding rock by line drilling can be loaded both thermally and mechanically. These studies will furnish basic data regarding jointed rock behavior under a variety of simulated repository conditions. The block will also provide a site for equipment and calibration.

Activities During the Reporting Period

No progress report was submitted for the period.

WBS 1.4.3

Project: NWTS Repository Equipment Development Plan

Principal Investigator: Battelle-Columbus (T. M. Trainer)

ONWI Project Manager: N. C. Henderson

Objective

The objectives for FY 81 and FY 82 are (1) to modify the Technology Development Plan for NWTS Repository Equipment in accord with Repository-IWG and DOE recommendations, (2) to prepare a draft Repository Equipment Test Plan, and (3) to assist the ONWI staff in preparing statements of work in accord with the Technology Development Plan.

Progress Reported Previously

This project was initiated in May, 1979, to assist the Engineering Development Department (EDD) of ONWI in developing procurement schedules for repository equipment. This effort resulted in a Preliminary Development Plan for Repository Equipment, a plan concerned primarily with repositories in salt. In FY 80 the plan was extended to include all NWTS repository processes and equipment requiring technology development. The FY 80 effort resulted in a final draft, Technology Development Plan for NWTS Repository Equipment, and a topical draft, Equipment Technology Development Required for a Nuclear Waste Geologic Repository.

Activities During the Reporting Period

The Technology Development Plan for NWTS Repository Equipment is currently being reviewed by NWTS personnel and the DOE.

The Repository Equipment Test Plan was identified by ONWI as being a comprehensive testing effort to ensure that the waste handling equipment installed at the repository will function according to predicted availability goals. This testing activity will be the mechanism to test all critical waste-handling and excavation equipment and to identify the need for changes prior to the actual installation or use of this equipment at the repository. Technical assistance in preparing the statements of work was also identified by ONWI as a pressing need.

Work during the first quarter of FY 81 resulted in the following accomplishments on the three project tasks.

**Preparation of a Draft
Repository Equipment Test Plan**

Available literature was reviewed concerning repository development and operational schedules to identify the different stages of testing appropriate for repository equipment. Preliminary testing schedules were developed for major types of equipment. The extent and purpose of the test plan were defined more clearly.

Preparation of Draft Statements of Work for Equipment Technology Development

Detailed consideration was given to the possible preparation of six statements of work. Through meetings with related personnel, alternative ways were found to obtain the information anticipated for four of the projects, and preparation of these statements of work was cancelled. Two draft statements of work were prepared: (1) Study of Equipment Alternatives for the Subsurface Handling of Canistered Nuclear Waste, and (2) Reliability and Maintainability Analysis of a Canistered Waste Handling System for a Nuclear Geologic Repository.

Future Activities

Second quarter activities on the three tasks are expected to be as follows: (1) meetings will be held with the Technology Development Plan reviewers; (2) the Repository Equipment Test Plan outline will be developed; and (3) a statement of work will be prepared on emplacement hole drilling.

WBS 1.4.3

Project: Engineering Design Criteria Methodology Development

Principal Investigator: Los Alamos Technical Associates, Inc. (R. J. Kingsbury)

ONWI Project Manager: N. C. Henderson

Objective

The objective of this study has been to develop a methodology and format which will be used to aid in preparing engineering design criteria and specifications for equipment used at a nuclear waste disposal facility.

Progress Reported Previously

The review, analyses, and characterization of the repository conceptual design criteria for NWTs Repositories Nos. 1 and 2 were completed, and a structured functional flow diagram was developed. This diagram is the foundation for the criteria development methodology.

Activities During the Reporting Period

A methodology has been developed to aid in preparing design criteria for nuclear waste repository equipment. The purpose of the methodology is to provide a comprehensive and systematic approach to be followed in preparing the initial equipment design criteria, updating the criteria as repository design progresses from conceptual through detailed design, and determining which equipment will be affected by repository design changes.

The methodology can best be visualized as a set of tools and comprehensive instructions for use in the development of equipment design criteria. The major elements of the methodology are:

- Functional flow diagrams for the waste receiving, storage, transfer, emplacement, and retrieval functions in a nuclear waste repository;
- A data base consisting of titles, sequence numbers, and descriptions for each function on the functional flow diagram and master checklists of equipment design requirements organized in a standard format; and
- A computer program that allows the user to generate specific repository equipment design criteria by modifying the functions and associated equipment design requirements in the data base to reflect the specific repository under development.

The functions, the functional flow diagrams, and the master checklists of equipment design requirements in the methodology were derived from a detailed review and systematic analysis of existing conceptual design studies for repositories.

The computer program, which serves as the methodology's data collection, organization, and management tool, was developed to operate on the ONWI computer system. The computer program and a separate, two-volume user's manual have been delivered to ONWI.

Future Activities

This project has been completed.

WBS 1.4.3

Project: Equipment Reliability and Maintainability Methodology

Principal Investigator: General Atomic Company (D. D. Orvis)

ONWI Project Manager: N. C. Henderson

Objective

The objective is to synthesize a methodology for equipment reliability, availability, and maintainability (RAM) analysis that can be used to establish criteria for equipment in an NWTS facility to ensure that a repository operates with high availability. This project involves study of the RAM requirements of reference facility designs and the available analysis methodologies, and then adapts those judged to be most appropriate for applicability to a waste repository. The methodology will be developed into a guidebook that can be used by ONWI, repository contractors, design personnel, and RAM specialists. The selected methodologies are being demonstrated through evaluation of selected portions of the Conceptual Reference Repository Design.

Progress Reported Previously

This project began on 24 March 1980. The Conceptual Reference Repository Description (CRRD) documents and descriptions of other conceptual designs were reviewed to gain familiarity with the concepts and operational requirements. Various RAM methodologies were reviewed for general applicability to the NWTS repository design effort. Those judged applicable were tailored to NWTS program needs and included in an initial Guidebook draft that was submitted to ONWI in July, 1980. Following ONWI review, more emphasis was placed on qualitative analysis and design guidance in addition to refining the quantitative methodology.

In refining the methodology, the following examples of application of RAM methodology to CRRD systems were developed: failure mode and effect analysis (FMEA), reliability block diagram (RBD), fault-tree analysis (FTA), and maintainability studies. Since RAM data based on operating records for specific equipment and equipment in a specific repository environment are not yet available, where possible, data from operations of similar equipment in different environments were used after adjustment for such differences. Alternatively, failure rate or repair time values were developed on the basis of expert opinion. Bayes decision rule was applied to combine data from several sources while maintaining information on the uncertainties through the use of a computer code named BROLS, developed for this purpose.

Activities During the Reporting Period

Whereas the principal activities had previously concentrated on developing sample applications of methodology to CRRD systems, the principal activity in this quarter centered on the RAM Guidebook. Its organization went through several stages in order to develop the proper sequence of presentation and level of detail. In the final draft form, the Guidebook consists of two volumes.

Volume I presents the overall RAM methodology recommended for the NWTS program. It begins with programmatic considerations on how and why to use the Guidebook. Chapter 1 describes the respective roles for program management, repository designers, and RAM specialists. Chapter 2 expresses the need for a RAM program and discusses how the program can be integrated with other repository development activities.

Chapter 3 expands upon RAM needs and concerns and introduces the conceptual design of a repository as represented by the CRRD. Chapter 4 develops the principles of designing for reliability and maintainability and is intended for use by design personnel to provide inherent reliability and maintainability in the equipment and systems for which they are responsible. Chapter 5 presents a structured, essentially qualitative RAM methodology for designers or other nonspecialists in RAM techniques: FMEA, RBDs, RAM checklists, and development of an availability improvement list (AIL). The latter is prepared by designers for management to help establish program priorities and resource allocation for RAM activities.

Chapters 6 and 7 introduce quantitative RAM methodology that is expected to be performed primarily by RAM specialists. Chapter 6 introduces the concept of reliability or availability goal setting and allocation of goals to subsystems and equipment. The preferred basis for developing the allocations is quantitative reliability and maintainability predictions based on failure frequency and repair time data for similar or identical items. However, an alternative process for allocation is presented that relies on the experience of cognizant design and maintenance personnel and RAM specialists to establish the relative reliability and maintainability characteristics of various systems and equipment. A computer program named ALOCAT was developed to perform goal allocation from either method.

Chapter 7 presents abbreviated descriptions of quantitative analysis methods for both reliability and maintainability. These include quantification of RBDs, FTAs, Markov models, event tree analysis, data search and synthesis, maintainability task analysis, and treatment of uncertainties. Examples of applications of the methodology are included in the chapter.

Volume II contains six appendices. Appendices A through D expand upon the quantitative methodology described in Chapters 6 and 7: allocation techniques in Appendix A; expanded treatment of reliability methods and additional examples in Appendix B; a summary of maintainability aspects in Appendix C; and a description of recommended computer programs in Appendix D. The latter includes the ALOCAT and BROLS codes, already mentioned, and the STADIC code for treating and compounding uncertainties in RAM data and models. These three programs were developed at General Atomic. Additional programs described in Appendix D are the MOCUS and KITT codes for use in FTA. These two codes are widely used and were developed elsewhere. Appendix E provides detailed reliability and maintainability checklists and RAM methodology worksheets for the convenience of users. Appendix F is a glossary of terms used in the Guidebook.

The second working draft of the Guidebook was completed and submitted to ONWI for review and comment.

Future Activities

This project is scheduled to be completed during the second quarter of FY 81. The remaining subtasks include resolution of ONWI comments on the Guidebook draft, publication of the final Guidebook, conversion of computer programs recommended in the Guidebook to the ONWI/Battelle computer system, and publication of the Final Report.

WBS 1.4.4**Project:** Directional Antenna**Principal Investigator:** Lawrence Livermore Laboratory (M. Buettner and L. Ramspott)**ONWI Project Manager:** J. O. Duguid**Objective**

The overall objective is to construct a directional antenna system, usable within a single borehole, for locating anomalies near the borehole. The goals are to:

- Show experimentally that miniature, directional antennas can be produced by enclosing the antenna elements in a sheath made of a ferrite or ferroelectric material
- Provide an assessment of the most appropriate sheath materials.

Progress Reported Previously

The shortening and directionality properties of a ferrite-sheath antenna were confirmed experimentally.

It was concluded that the antenna pattern for the ferrite-loaded dipole cannot be determined from time-domain measurements made using the LLL transient antenna range. This range has insufficient clear time for such a measurement.

Work began on predicting the radiation pattern for a dipole antenna loaded by a finite cylinder of a ferrite or ferroelectric material.

The preliminary assessment of which materials best meet the design criteria was reached. The essential points are described below.

1. For a given diameter cylinder, an antenna constructed of a ferroelectric such as barium titanate will show shortening and directionality at a much lower frequency (perhaps 60 MHz) than an antenna constructed of a ferrite (around 250 MHz). However the ferroelectric antenna has a much lower efficiency than its ferrite counterpart.
2. The ferrite antenna should be a good choice in low-loss geologic media, such as good quality rock salt or granite where radio frequency energy can travel reasonably large distances at frequencies around 200 MHz. The ferroelectric antenna would seem to be the choice for media where only lower frequency energy can travel large distances (though one must cope with the low efficiency).
3. The conclusions regarding the ferroelectric-sheath antenna were based solely on theoretical considerations since the actual material had not yet arrived.

Activities During the Reporting Period

An experiment was conducted at Gold Hill, Colorado, in October. The radiation pattern for the ferrite-loaded dipole antenna was measured in granite with promising results. The relative dielectric constant of granite near 200 MHz was not measured because of equipment failure, but it was estimated to be about 2. The input impedance of the antenna downhole was not measured because of equipment failure.

Mathematical difficulties have caused a change in approach to the problem of predicting the radiation pattern for a dipole antenna placed on a ferrite or ferroelectric cylinder. An alternative formulation has now been completed, and coding for the computer has begun.

Complex permittivity and permeability for the ferrite and ferroelectric specimens could not be determined at Colorado State because of problems with the measuring system.

Some of the analysis has been completed, and equipment is being gathered for a system at Livermore similar to that at Colorado State for determining complex permittivity and permeability.

Future Activities

The following activities are planned:

1. Computer coding for the ferrite cylinder problem will proceed, and some test cases will be examined.
2. Analysis of predicted system performance will begin.
3. Complex permittivity and permeability will be determined for the ferrite and ferroelectric specimens at Colorado State.
4. Analysis will continue toward the goal of building a system at LLL for measuring complex permittivity and permeability.
5. Input impedance for the ferrite-loaded dipole antenna will be measured in a borehole at Livermore.

WBS 1.4.4**Project:** Data Transmission Alternatives**Principal Investigator:** TRW, Inc. (P. Alexander)**ONWI Project Manager:** N. C. Henderson**Objective**

The objective of this project is to develop and analyze techniques to transmit, to the surface, sensor data collected within, or adjacent to, a subsurface nuclear waste repository without breaching the geologic integrity. The transmission system should be compatible with a wide variety of undefined measurement techniques and adaptable to the geologic media encountered in all candidate repositories. Four subobjectives support the overall objective:

1. Establish which performance requirements are most critical to the success of long-term monitoring systems.
2. Review the prototype long-wavelength wireless telemetry system, and systematically identify its strengths and weaknesses and those of its major subsystems.
3. Identify and conceptualize alternative data transmission ideas and similarly analyze their strengths and weaknesses.
4. Analyze and compare all candidate systems relative to the established critical performance requirements, recommending those systems indicating superior performance for long-term repository monitoring.

Progress Reported Previously

This project began on 16 June 1980. A program plan was submitted to ONWI the next month. It described the project objectives, the tasks to be undertaken to fulfill them, and the technical approach to be used while performing each task. No significant deviations from the approach described in the proposal were necessary.

The critical system performance requirements task was completed during the second quarter to the point of delivering an oral presentation of the results and submitting a draft report. Two more tasks were initiated: a review of the prototype long-wavelength wireless telemetry system and the identification and conceptualization of alternative data transmission concepts.

Activities During the Reporting Period**Procedures**

The prototype long-wavelength system was initiated by examining the design and any claimed performance characteristics. All assumptions used to define the performance claims were also reviewed. Many performance characteristics were derived.

Conceptualizing the alternative data transmission techniques started with identifying those that have been used in a wide variety of technical applications. No bounds were placed on the identification process initially. Each technique was then viewed in the context of a waste repository environment. Many operational difficulties became immediately apparent with each technique. As the difficulties in operating in that environment were addressed, the alternative concepts were developed into candidate systems.

Results

Written descriptions of the long-wavelength wireless prototype system and the alternative concepts have been prepared. One consistent format was used for each writeup:

- A brief concept description
- System components
- Concept strengths and weaknesses
- Schematic
- Performance characteristics.

These concept descriptions have been sufficiently drafted and edited to include all but the performance characteristics sections.

Future Activities

The project is expected to end on schedule during the next quarter. The remaining task to be completed is the comparison of candidate systems. A final report will be written.

WBS 1.4.4

Project: Monitor for Detecting Nuclear Waste Leakage in a Subsurface Repository

Principal Investigator: Lawrence Berkeley Laboratory (S. Klainer)

ONWI Project Manager: N. C. Henderson

Objective

The purpose of this research is to develop an ultrasensitive specific monitor for the actinide elements. The approach provides for two modes of analysis: coprecipitation enhanced fluorimetry (CEF) and remote fiber fluorimetry (RFF). The end result of this research will be to provide an "early warning system" and a long-term monitoring capability for incipient waste-package failure. Such instrumentation will also assist in the conduct of certain field experiments such as brine migration and nuclide migration.

Activities During the Reporting Period

No progress report was submitted for the period.

WBS 1.4.4

Project: Nuclear Quadrupole Resonance (NQR) Techniques to Measure Stress and Strain in Rocks

Principal Investigator: Lawrence Berkeley Laboratory (M. Klainer)

ONWI Project Manager: N.C. Henderson

Objective

The goal of this project is to develop a new technique for measuring underground stress. The measurement of the in situ state of stress in rocks is a challenging and important problem because it must be known to ensure the integrity and safety of underground openings. Present stress measurement techniques are costly, time-consuming, often unrepeatable, and difficult to carry out. This program uses the microscopic response of crystalline materials to stress, particularly changes in the chemical bonds that increase the electric field gradient (EFG) tensor. The EFG is observed using nuclear quadrupole resonance (NQR) or a double resonance technique involving NQR and nuclear magnetic resonance (NMR). The stress tensor can be determined from the NQR frequencies observed in the rocks surrounding a borehole. The first phase of the project is focused on stress measurements in rock salt, and later aluminum-containing minerals will be investigated. Faster and more reliable stress measurements will be valuable in the site-selection process leading to an underground nuclear waste repository.

Progress Reported Previously

The project began in June 1980. The stress-induced quadrupole frequencies in NaCl lie in the 1 to 10 kHz region for stresses at the 10 to 100 atm level. A double resonance technique, in which the sodium NMR lines in a ca. 1,000 gauss magnetic field are used to detect the low-frequency chlorine lines, will be used first. Calculations show that there will be adequate sensitivity and signal-to-noise ratio. Resolution should be a few atmospheres or better. The three-dimensional character of the stress will be found by rotating the instrument in the borehole and using the borehole perturbation of the stress field to advantage in the mapping.

Activities During the Reporting Period

A relatively small fraction of the total effort has been devoted to the search for ^{27}Al resonances in kyanite and spodumene, so far without success. The direct observation of the free induction decay and the technique of spin echo were used in all cases with signal averaging.

The NMR data of kyanite show that one of the transition frequencies should appear at 2.728 MHz, which is just around ^{14}N resonance for NaNO_2 at room temperature. No signal could be observed by varying the different experimental parameters like pulse width, pulse interval, and RF power. Attempts were also made at the temperature of liquid nitrogen.

For spodumene, the NQR frequencies are known from an earlier experiment with a single crystal. The frequencies for the two transitions are around 750 kHz, which necessitates the use of a sample coil having a large number of turns.

A number of details and different methods of performing double resonance experiments have been studied with a view to identifying the approaches with the highest probability of success in sodium chloride. Double resonance by continuous coupling appears to be the most favorable method at present.

WBS 1.4.5**Project:** Repository Sealing Project**Principal Investigator:** D'Appolonia Consulting Engineers, Inc. (D. E. Stephenson)**ONWI Project Manager:** F. L. Burns**Objective**

The primary objectives of the penetration sealing program are (1) to develop seal designs to fulfill the design goals of providing safe seals at a reasonable cost, (2) to provide seal designs on schedule to satisfy overall repository licensing requirements, and (3) to provide coordination and interfacing with the other ongoing repository sealing studies being performed in the laboratory and at ONWI field sites.

Progress Reported Previously

This project was initiated on 6 November 1978. Two reports, ONWI-3, *Development of Plan and Approach for Borehole Plugging Field Testing*, and ONWI-15, *The Status of Borehole Plugging and Shaft Sealing for Geologic Isolation of Radioactive Waste*, have resulted from the program. ONWI-55, *Repository Sealing Design Approach—1979*, describes initial design concepts for borehole plugs and defines a design philosophy. A draft QA manual for the repository sealing program was completed and forwarded to ONWI for review. Liaison with the field activities at the WIPP site continued. The geochemical needs of the repository sealing program were reviewed, and a preliminary outline of the requirements was formulated and reported in ONWI-108. Several initial seal designs were analyzed with numerical techniques to determine their effect on fluid flow through the seal zone and host rock. A report on the differences in borehole shaft and tunnel sealing requirements, ONWI-90, was completed. A review of the draft NRC regulations for HLW disposal was performed for ONWI.

Activities During the Reporting Period

A preliminary design concept in the form of a working design for sealing a repository at the Los Medaños site was developed during the quarter. The design is intended to stimulate comment and thought and will be revised periodically until the conceptual sealing design for the site has been completed in August, 1982. The present working design for shaft and tunnels is shown in Figure 1. Continued refinement of the design concept is being performed by numerical modeling. The design philosophy for this working design includes:

- A unified system including shaft seals and tunnel backfill
- A module to prevent inflow of water into the repository
- A module to prevent the escape of radionuclides to the biosphere

FIGURE 1. TUNNEL-SHAFT SEAL SYSTEM SCHEMATIC

- The use of relatively well-known materials and simple shapes
- The use of backup components for redundancy.

The load conditions and systems properties were estimated for the site and influenced the design concept.

Six priority characteristics were established which should be investigated by a field testing program in FY 81: (1) fracture distribution, (2) fracture characterizations, (3) permeability measurements of disturbed and undisturbed zones in low-permeability rocks, (4) evaluation of the extent of the disturbed zone around penetrations, (5) sidewall characteristics, and (6) ground-water geochemistry. The recommended penetration sealing field test program for FY 81 is designed to provide data and develop techniques to satisfy design requirements related to these characteristics. Recommendations for obtaining these data on a site-specific basis were developed based upon the ONWI milestone schedule for conceptual and final sealing designs. Where suitable penetrations or facilities are not available at a potential repository site, generic field testing was recommended at available sites.

Materials research goals for repository sealing design was reviewed by all participants in the ONWI program working on seal design and materials. The working design for the Los Medanos site was reviewed, and a consensus on which materials may perform in the sealing system and should be investigated was agreed upon. The properties of the materials to be used as a measure of seal performance and applicability as related to design were determined.

Review studies performed on some of the materials proposed in ONWI-108 for repository sealing are in progress. These include clays, polymer cements, and bituminous materials. The polymer cement report concluded that these materials, currently under investigation for use in geothermal systems, may be applied to sealing to reduce fluid flow, although microbiological degradation may be a problem area. The clay study will conclude that clays can be expected to provide seal functions and probably meet longevity requirements. The bitumen report has not progressed sufficiently to formulate conclusions.

Future Activities

During the next quarter, it is planned to complete the study of current shaft construction and shaft seal design and to evaluate testing needs for shaft seals related to repository sealing. An interim report on the working design of the sealing system for the Los Medanos site will be completed. The investigation of instrumentation needs for repository sealing will be initiated during the latter part of the quarter.

REFERENCES

An Evaluation of Generic Similarities and Differences in Requirements for Sealing Shafts, Tunnels, and Boreholes, 1980. ONWI-90, Office of Nuclear Waste Isolation, Battelle Memorial Institute, Columbus, OH.

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The Status of Borehole Plugging and Shaft Sealing for Geologic Isolation of Radioactive Waste, 1979. ONWI-15, Office of Nuclear Waste Isolation, Battelle Memorial Institute, Columbus, OH.

WBS 1.4.5

Project: Borehole Plugging Program

Principal Investigator: Sandia National Laboratories (C. L. Christensen)

ONWI Project Manager: F. L. Burns

Objective

The objectives of this program are:

1. To develop, test, and evaluate candidate cementitious grouts and other materials for use in and support of the laboratory and field-testing program; to assist in developing and implementing a geochemical program to ascertain and verify the long-term stability and suitability of cementitious and other materials as wellbore plugging materials
2. To assess, develop, and modify appropriate instrumentation techniques to obtain the data required to verify plug performance
3. To provide an in situ environment for emplacing and evaluating potential plug designs.

Progress Reported Previously

The current Borehole Plugging Program (BHP) was initiated in September, 1978, with full-scale operation under way by February, 1979. The program has evolved into four major tasks: management, materials, instrumentation, and field test. Materials development centers around cementitious plugging materials with both fresh-water and brine mixes, along with fly ash and other additives to improve permeability and emplacement characteristics; instrumentation development assesses program needs and provides resources as required; and the field test implements the in situ plug emplacement, testing, and verification as required. The current field test is the Bell Canyon Test (BCT). The Potash Core Test (PCT) is scheduled to begin.

Previous work indicates that cementitious materials should provide adequate plugs; existing technology coupled with economics make them attractive. One useful result showed that push-out strength tests are inadequate to assess the permeability of the interfaces between the grout and the rock. Heretofore bond strength was considered indicative of plug adequacy, but a strong bond does not necessarily result in low interface permeability. The type of mix water is important. Fresh-water mixes alleviated the permeability problem of brine mixes at the interface. Indications are that in nonsalt emplacement, the impact of brine mix water must be carefully assessed. Testing of this hypothesis is under way at the U.S. Army Corps of Engineers, WES, Vicksburg, MS. This testing is directed at determining the acceptable salinity which should be permitted in a given plugging application.

A draft report on the BCT-1FF plug was issued for review. Updates on BCT results and progress have been forwarded to ONWI.

Activities During the Reporting Period

Management and Design

Negotiations with Systems, Science, and Software have been completed for the design, construction, and laboratory testing of the fluid permeability measurement system. Once the system has been adequately tested in the laboratory, plans for a full-scale in situ evaluation will be initiated.

Contact has been made with personnel at Woodward-Clyde regarding use of the PCT core barrel in the recovery of the Kubat well plugs.

Materials

Testing continued on the original five laboratory mixes at the four-year interval; on the U.S. Energy Research and Development Administration's 10 samples at the three-year interval; and on the BCT-1FF samples at the one-year interval. This testing is a continuation of the program to develop a history of aging effects with respect to specimen length and volume changes, porosity, density, dynamic modulus, compressional wave velocity, and permeability. Increased emphasis has been placed on X-ray diffraction and scanning electron microscopy techniques to complement the physical tests noted above.

Field Test

The emphasis of this subtask is on preparing for the conduct of the PCT. The operation plan for the PCT has been completed and reviewed. Rig compatibility requirements with the WIPP-14 operation have been determined. All identified core-handling equipment has been designed and is being fabricated. Site preparation is in progress; the pad for the core-handling building is poured, and the P-13 cellar has been excavated to the 20-foot depth.

Preparations for testing BCT plug ONEX have been deferred pending completion of the PCT.

Future Activities

No new activities have been planned. Activities that require continuation include BCT plug ONEX testing, program test planning, and those activities required to support repository development in southeastern New Mexico.

WBS 1.4.5

Project: Seal Materials Selection and Longevity

Principal Investigator: Pennsylvania State University (D. M. Roy)

ONWI Project Manager: F. L. Burns

Objective

This project is oriented toward the development of improved technology in cements and cementitious materials in conjunction with their specific role in borehole plugging, grouting, and shaft sealing of various natural repositories for radioactive waste isolation.

The problems of processing, emplacement, reliability, and long-term stability are addressed through: (1) establishing the effects of compositional variables upon the rheological properties, (2) investigating volume stability, (3) modifying the nature of cement hydrate, (4) investigating the interface/bond between cement and host rock, (5) studying accelerated reaction kinetics, and (6) identifying various mechanical/physical properties as determinants in predicting long-term performance.

Activities During the Reporting Period

No progress report was submitted for the period.

WBS 1.4.5**Project:** Seal Materials Performance Limits and Longevity**Principal Investigator:** Corps of Engineers, Waterways Experiment Station (K. Mather)**ONWI Project Manager:** F. L. Burns**Objective**

The Structures Laboratory will optimize proportions for grouts to be used in tests at AEC-7 and similar geologic settings. Cements, pozzolans, and admixtures with fresh water or water containing salts appropriate to render the grout compatible with appropriate physical, mechanical, chemical, and thermal properties of the wall rock will be considered. This will require characterization of wall rock, grouts, and ultimate hydration products. Also investigated will be other grout mixtures for different geologic settings using different materials and expansive systems. Precision and accuracy data on methods for characterization and testing will be developed.

Progress Reported Previously

Work began on 2 January 1980. Specimens were cast in the task of optimizing the BCT-1FF grout used in the Bell Canyon Test (BCT). The joint PSU/WES experimentation to develop precision and accuracy data began.

Activities During the Reporting Period**Optimization of the BCT-1FF Grout**

Work began on evaluating the nine mixtures made in optimizing the Bell Canyon Grout, BCT-1FF. A mixture proportioned by the Materials Research Laboratory (MRL) of Pennsylvania State University was prepared at the Structures Laboratory (SL), using materials from the same sources and in most cases from the same or closely related samples of pozzolan, portland cement, plaster of paris, high-range water reducer, and defoamer. The salt used at the MRL was reagent grade and that used at the SL was Carey's Coarse Granulated Salt from a supply on hand that has been used to make brines containing 30 and 36 percent by mass NaCl used as mixing water in other salt-water grouts, and in synthesized brines for several uses. The water used to make the grouts and in the permeability tests was deionized. Mixing and casting cubes, bars, and vials of grout at the SL followed ASTM procedures except for (1) a modification of the ASTM C 305 mixing procedure, and (2) the omission of aggregate from the mixture used to make ASTM C 109 cubes, bars 1 inch by 1 inch in cross section and 11-1/8 inches long, restrained expansion bars, and vials of grout to be examined by X-ray diffraction and scanning electron microscopy (SEM). Because the scanning electron microscope was out of order when the earliest SEM examination was scheduled, the specimen was examined and micrographs were made after the specimen had been held in methanol in a freezer for a few days. X-ray diffraction examinations and determinations of dynamic modulus of elasticity and compressive

strength were made as scheduled. The SL examined the within-laboratory and between-laboratory standard deviations or coefficients of variation for all ASTM test methods used which have precision statements permitting test results to be compared within and between laboratories.

Compressive strength tests and measurements of restrained expansion bars of the nine mixtures cast in September, 1980, were performed toward optimizing the BCT-1FF grout according to ASTM C 109 and C 806. Examinations of progressively older specimens were made by X-ray diffraction.

Consistency in the BCT-1FF grout was controlled by using a flow cone, checking against the efflux time of deionized water, and adding additional experimental water reducer (AD-627) when the efflux time for the grout exceeded the time expected to be satisfactory. The more water reducer used, the more it tended to delay the setting time. Strengths and restrained expansions have been normal for the ages reached, except for one grout with low early strengths and high restrained expansion in a mixture containing a shrinkage-compensating cement with fly ash and added plaster.

The tendency of the experimental water reducer to delay setting of the grout mixtures supports other evidence that the water reducer is a high-range water reducer. The high expansion of the mixture containing a shrinkage-compensating cement with fly ash and added plaster will be investigated. Other mixtures show strength development and expansion comparable with the Bell Canyon laboratory and field grouts. The highest strengths, so far, are produced using one Class H cement which is coarsely ground, with about 30 percent by volume replacement of cement by a class C (or low total of $\text{SiO}_2 + \text{Al}_2\text{O}_3 + \text{Fe}_2\text{O}_3$, and fairly high CaO content) fly ash. Subsequent results at greater ages using a Class F fly ash should be significant and interesting.

Examination of Laboratory-Cast and Field-Cast Grouts

When the laboratory-cast and field-cast grouts from the proportioning studies, ERDA-10 plugs, and Bell Canyon plugs underwent physical tests and examinations by SEM and X-ray diffraction, their properties reflected reasonable developments from earlier tests.

With not all results in, it appeared that the results at the two laboratories in the PSU/WES cooperative work fell well within the between-laboratories standard deviations or coefficients of variation for the ASTM test methods used. The nonstandardized methods also agreed well. Only two apparently significant anomalies between the results of the two laboratories were found; these are being double-checked, and mistakes in calculation have been found.

No activity in field surveillance of grouting, field testing of grout, or preparation of specimens from grout placed in any field site was scheduled, but it was decided to continue the long-term expansion tests, including permeability measurements and determinations of dynamic modulus of elasticity and unconfined compressive strength. Progressively older laboratory mixtures developed during planning for field tests at ERDA-10 and the Bell Canyon tests, as well as field-cast specimens from these projects, will be used. Some of these are approaching 5 years of age and have been investigated at regular intervals. Permeability specimens from the PSU/WES cooperative study and from the optimization of BCT-1FF grout are being tested with a new permeability apparatus that holds six specimens.

The petrography (SEM) and X-ray diffraction examinations continued on specimens from ERDA-10 and the Bell Canyon tests.

The extension in time of the tests and examinations of the laboratory-cast and field-cast grouts reveal a consistency in their behavior at earlier ages and as they cure. Since some of these specimens are approaching 5 years of age and have been cured and tested at regular intervals in controlled curing, they are believed to represent some of the oldest and most thoroughly documented laboratory-cured, field-cast, and laboratory-cast specimens of grouts proportioned and tested for use in borehole plugging in nuclear waste repositories that are available in this country.

Future Activities

A 65-year-old foundation grout from the Troy, NY, Lock and Dam, that is still in good condition after having been in contact with steel for more than one-half century, will be studied.

A study of two generations of simulated borehole specimens that were created in connection with the BCT should advance the understanding of how cracks develop between grout and surrounding anhydrite.

The effects of curing BCT-1FF grout in several temperatures and pressures will be reported.

Work will begin on the stability of ettringite and tetracalcium aluminate 2-chloride 10-dihydrate in the range of 66 to 100 C, for various curing times.

WBS 1.4.5**Project:** Wellbore Damage Zone Determination**Principal Investigator:** Terra Tek, Inc. (M. D. Voegelé)**ONWI Project Manager:** F. L. Burns**Objective**

The objective of this program is to obtain experimental data on the wellbore damage zone caused by full-scale drilling operations. The extent and characteristics of the damage zone will be evaluated on five rock types. The drilling will be performed in the Terra Tek Drilling Research Laboratory (DRL) using as close as possible the same drilling parameters, bits, drilling fluids, and fluid pressures as those used in the field, and with the rock sample subjected to in situ environmental conditions. The drilling damage will be assessed by making permeability measurements before and after drilling and by dye-penetrant tests. A full-scale laboratory plug leakage test will also be performed. This effort is in direct support of ONWI's repository sealing program and is interactive with an ONWI borehole plug design contractor conducting concurrent modeling studies.

Progress Reported Previously

This project began on 1 August 1980. During the first reporting period, the rock type #1 (anhydrite) was obtained and the seven drill test samples were prepared. Drilling tests were completed on six of the anhydrite samples (all except the plug test sample).

Activities During the Reporting Period

The project was extended to include a total of five rock types: (1) anhydrite, (2) salt, (3) granite, (4) basalt, and (5) tuff. The revised test matrix is shown in Table 1.

Progress made in obtaining the samples is as follows:

1. The first attempt to obtain salt samples at Avery Island, Louisiana, was unsuccessful. Modifications to the equipment are under way, and it is anticipated that drilling test samples will be obtained soon.
2. The block of charcoal gray granite (Cold Springs Granite) has arrived for testing. It has abundant dark-gray hornblende and biotite-rich inclusions, as well as uniform texture; it is fairly isotropic. Two of the drilling test samples have been prepared from the block.
3. A site on the Columbia River near the John Day Dam in the state of Washington has been selected as a source for the basalt test samples.
4. The search for a site to obtain the tuff test samples was initiated. Communications with Sandia National Laboratory personnel indicated that the G-tunnel complex at the Nevada Test Site would *not* be a good candidate site for large cores because of the high-density jointing. Test samples of tuff from the Yucca Mountain area of the NTS

TABLE 1. TEST MATRIX(a)

Rock 1: Anhydrite							Rock 2: Salt						Rock 3: Granite						Rock 4: Basalt						Rock 5: Tuff					
Bit	Drilling Fluid	Drilling Test	Damage Pulse Test	Plug Test	Permeability		Drilling Test	Pulse Test	Plug Test	Permeability		Drilling Test	Pulse Test	Plug Test	Permeability		Drilling Test	Pulse Test	Plug Test	Permeability		Drilling Test	Pulse Test	Plug Test	Permeability					
					Before	After				Before	After				Before	After				Before	After				Before	After				
	Air	X			XX	XXXXX	X			X	XXX	X			X	XXX	X			X	XXX	X			X	XXX				
Roller-Cone	Mud	X X	X	X	XX X	XXXXX XXXXXX	X			X	XXX	X			X	XXX	X			X	XXX	X			X	XXX				
	Mud and Cleaning Fluid	X			XX	XXXXX	X			X	XX	X				XX	X			X	XX	X				XX				
Diamond	Water	X			X	XXXXX																								
	Mud	X			X	XXXXX																								
	Mud and Cleaning Fluid	X			X	XXXXX																								
7 1 1 10 36							3	0	0	3	8	3	0	0	2	8	3	0	0	3	8	3	0	0	2	8				

(a) Total Tests: Drilling Tests 19
Pulse Tests 1
Plug Tests 1
Permeability Tests 88
Dye-Penetrant Tests 60

would be ideal for the drilling tests. However, U.S. Geological Survey information indicated that there is now known outcropping with accessibility sufficient for transporting equipment or large samples.

While the anhydrite samples were in the DRL pressure vessel, several permeability tests were attempted. The results of the only completely satisfactory permeability tests indicated a permeability coefficient of approximately 10^{-7} darcy (10^{-16} cm²). This data point was obtained by a pulse test with nitrogen as the permeant in an air-drilled sample. Permeability tests using water and mud were unsuccessful primarily because of equipment problems. They were valid, however, from the standpoint of indicating equipment modifications and indicating proper interpretive techniques to be used in subsequent tests. Additionally, fluorescent dye was added to the drilling mud and the pertinent specimens were examined following drilling and injection testing. The results of the investigations indicated that flow through the rock mass occurs primarily through fractures. The implication here is that the calculated permeability coefficient cannot be assuredly assumed to be a true permeability coefficient; none the less, it does serve as a valid basis of correlation for large-scale specimen comparison.

Future Activities

1. A site will be selected, and test samples of tuff will be obtained and prepared for drilling during the next quarter.
2. The base-line permeability of rocks #1 (anhydrite), #2 (salt), and #3 (granite) and the after-drilling permeability measurements on the anhydrite (except the plug flow test sample) are expected to be completed.
3. The plug flow test sample will be drilled, the grout plug will be poured, and the curing under pressure will begin. Rock samples #2 and #3 should also be drilled during this period.
4. The dye-penetrant tests will be completed on all available samples.

WBS 1.4.6

Project: Extended Study of Horizontal Burial

Principal Investigator: Bechtel National, Inc. (J. M. O'Donnell)

ONWI Project Manager: F. Djahanguiri

Objective

The objective of this engineering study is to compare the effects of horizontal emplacement of spent fuel in canisters in a repository layout, similar to the NWTS Conceptual Reference Repository Description (CRRD) in dome salt. The CRRD design was based on vertically emplacing a single canister in a predrilled hole. This study is based on horizontally emplacing a single canister in a predrilled hole. The comparison is to be made on both a technical and an economic basis. A relative assessment will also be made of horizontal versus vertical emplacement for host rock media other than salt. The study is divided into the following technical tasks:

- Determine the impact of horizontal burial on the CRRD
- Assess the impact of horizontal burial on a basis of cost, safety, and R&D effort required
- Assess relative merits of horizontal burial in other media.

Progress Reported Previously

This project began on 15 October 1980. This is the first quarterly technical progress report.

Activities During the Reporting Period**Revisions to the CRRD Specified
for Horizontal Emplacement**

All CRRD systems affected by horizontal burial were reviewed, and the following were revised and provided the primary basis for the comparison:

- Canister transfer cask for emplacement and retrieval
- Drilling and sleeving equipment for emplacement holes
- Equipment for transporting the casks and emplacing and retrieving the canisters
- Canister arrangement and spacing within the storage rooms
- Storage room and drift sizing

- Repository layout
- Ventilation requirements (technical comparison only)
- Salt handling system and requirements.

Determining the Impact of Horizontal Emplacement

The principal impact of horizontal emplacement is the reduction in the amount of underground excavation required. Preliminary results indicate an approximate 40 to 45 percent reduction in excavation, based on the increased number of canisters that can be stored in a room, the ability to use many of the entryways for horizontal storage, and the use of storage rooms with a 16 percent smaller cross-sectional area.

Another impact of horizontal emplacement is simplification of the underground equipment used to transport and emplace the canisters.

Because of the reduced excavation required for horizontal emplacement, the size and capacity of some of the underground auxiliary systems, such as ventilation, salt mining, and transporting, could be reduced.

The storage room configuration specified for horizontal emplacement would be lower and wider than that specified for the CRRD. In dome salt, the lower, wider configuration could be more stable.

Assessing the Revisions of Horizontal Emplacement

The CRRD cost estimate (for vertical emplacement) was reviewed primarily for the items listed above as a basis for a cost comparison with the estimate performed for the underground arrangement and equipment necessary to accommodate horizontal emplacement. The total constructed cost estimate was based on the first 5 years of project site development. The preliminary results indicated a potential total estimated constructed cost savings for the horizontal arrangement of \$55 million. An additional average cost savings of \$25 million per year for annual operating and maintenance costs could be realized for the horizontal arrangement for the remaining 31 years of plant life.

From a mining and radiological standpoint, the overall safety of the horizontal emplacement method appears equal or superior to vertical emplacement.

For the existing equipment and equipment proposed for horizontal emplacement (using the single canister predrilled hole concept), the research and development program required to demonstrate the concept would not be excessive, mainly equipment development and testing of modifications to existing equipment or components.

Assessing the Relative Merits of Horizontal Versus Vertical Emplacement in Media Other Than Salt

Preliminary results indicate that, for the host rocks considered (basalt, granite, schist, sandstone, shale, limestone/dolomite, and alluvial fan):

- The horizontal method represents an alternative to vertical emplacement that, in certain circumstances, may be the more favorable method to promote canister isolation and storage integrity.
- The influence of the geotechnical parameters studied on the choice of horizontal versus vertical emplacement of canisters is almost always site-specific in all of the media considered.

Future Activities

The final technical report for the tasks described above will be completed in the next quarter.

WBS 1.4.6

Project: Technology Development Studies for Geologic Disposal: Effects of Earthquakes on Subsurface Facilities

Principal Investigator: Savannah River Laboratory (SRL) (I. W. Marine)

ONWI Project Manager: M. R. Wigley

Objective

The project objective is to evaluate the potential for seismic disruption of underground facilities. The program is divided into the following tasks:

- Review the existing knowledge to document the damage or nondamage to mines, tunnels, wells, and other underground facilities due to earthquakes, and to evaluate the significance of these data. The report documenting this damage and nondamage was published by the Savannah River Laboratory as USDOE Report DP-1513, *Earthquake Damage to Underground Facilities*.
- Perform a detailed mathematical analysis of displacements in the vicinity of faults.
- Through the use of numerical techniques, perform a generic study to determine the effect of an earthquake on underground facilities, with emphasis on how and to what extent seismic waves might affect or create fractures which could permit increased water circulation in the rock surrounding a repository.

Activities During the Reporting Period

No progress report was submitted for the period.

WBS 1.4.6

Project: ONWI Support Program: Effects of Water in Salt Repositories

Principal Investigator: Oak Ridge National Laboratory (C. F. Baes, Jr.)

ONWI Project Manager: J. F. Kircher

Objective

The objective of this program is to predict from experimental and modeling studies those consequences of the presence or the ingress of water to a salt repository that might compromise the integrity of the repository.

Progress Reported Previously

Following initial studies of salt deposition onto beds of crystals from flowing brine in a negative temperature gradient, studies of the isothermal consolidation of salt beds under uniaxial stress in flowing brine were begun. The initial results, obtained in a 1.27-cm column with particle size (z) in the range 0.012 to 0.035 cm and applied uniaxial stresses (σ) up to 155 bars, revealed that the void fraction decreased with an approximate linear dependence on the logarithm of the product $\sigma t/z^2$, where t is the time. The permeability of these beds was found to decrease considerably faster with void fraction than is predicted by the Blake-Kozeny equation. A simple model was evolved for salt consolidation, based on the increase of salt solubility with pressure, which accounted approximately for the dependence of consolidation on the stress and particle size, but failed to account for the time dependence.

The consolidation results from the 1.27-cm-ID column gave evidence of a wall effect, that is, the void fraction was found to decrease with distance from the (top) end of the column where the stress was applied. Consequently, a larger (2.54-cm-ID) column was built for subsequent measurements. These indicated that, while the wall effect was still present, it was considerably reduced. Examination of the compacted larger columns revealed an approximately linear increase in the void fraction down the column.

A series of salt consolidation runs was completed in the larger apparatus in which the crystal size, the applied stress, and the length of the column were systematically varied. Runs were made in which the pure NaCl brine was replaced by a MgCl_2 -rich brine (in which NaCl is much less soluble), by dodecane, and by air. The rate of consolidation was found to decrease with each replacement. When the temperature, usually about 20 C, was increased (in the range 20 to 85 C), the rate of consolidation in brine increased, but the effect was not very large. Finally, the effect of particle-size distribution was examined. When the narrow size range usually used was broadened, the principal effect was to lower the initial void fraction, but there was no great change in the time dependence of further compaction.

It was found that the data for all runs could be represented, usually within experimental error, by a three-parameter equation of the form

$$\phi^0 = a - b \ln(c + t), \quad (1)$$

where ϕ^0 is the average void fraction for the column. The dependence of the parameters a , b , and c on the particle size, the applied stress, and other conditions could not be assigned reliably, however, because the parameters covaried so strongly with one another.

Microscopic examination of polished sections of consolidated columns indicated considerable alteration of crystal shape and the entrapment of brine inclusions at the intersection of grain boundaries.

An apparatus was built to measure the permeability of a single grain boundary between two NaCl crystals to flowing brine as a function of the applied stress normal to the boundary and the time. Initial results indicated that, at constant stress, log permeability decreases linearly with log time.

An apparatus was designed to study permeability and brine migration in 10-cm-OD halite core specimens under uniaxial stress and in a temperature gradient.

Activities During the Reporting Period

Salt Consolidation

During the reporting period, an additional consolidation run was made to study the effect of particle size distribution. Five runs made so far under the same conditions are compared in Figure 1.

A generalized treatment of the consolidation of salt in brine has been adapted from behavior reported in the literature for the sintering of metal and ceramic powders. Coble (1961a) concluded from a highly idealized model that the rate at which the void fraction (ϕ) decreased during sintering should be inversely proportional to the cube of the particle size z :

$$d\phi/dt = -A/z^3. \quad (2)$$

He also observed experimentally that the cube of the average particle size grew linearly with time during much of the sintering process (2):

$$\bar{z}(t)^3 = \bar{z}(0)^3 + Bt. \quad (3)$$

Assuming that z in Equation (1) can be replaced by the average particle size $\bar{z}$, these equations can be combined and integrated to give:

$$\phi(t) = \phi(0) - (A/B) \ln[1 + B(t/\bar{z}(0)^3)]. \quad (4)$$

Measurements of the decrease in the observed void fraction ϕ^0 during the isothermal consolidation of salt can be represented quite well by Equation (1) above. A comparison of Equations (1) and (4) show that they are equivalent if:

FIGURE 1. THE CONSOLIDATING COLUMNS OF SALT (2.45 cm IN DIAMETER) IN BRINE FOR MIXTURES OF PARTICLES OF VARIOUS SIZE RANGES

$$\phi(0) = a - b \ln c$$

$$A = b\bar{z}(0)^3/c$$

$$B = \bar{z}(0)^3/c.$$

The dependence on conditions of the values of $\phi(0)$, A , and B given by these relationships has been examined. As expected, the initial void fraction $\phi(0)$ seemed to decrease with increasing applied stress, the dependence on stress being approximately linear. The quantity A/B , equal to the limiting negative slope b of plots of ϕ vs $\ln t$, was found to increase approximately linearly with increasing applied stress.

These correlations were by no means perfect because the local stress at various points in a column is less than the applied stress by an amount depending on the magnitude of the wall effect, and for the same reason the local void fraction (ϕ) differs from the observed void fraction (ϕ^0). These correlations do suggest, however, that for the isothermal consolidation of salt in brine, a good representation of the time dependence of the local void fraction would be given by Equation (3), assuming that $\phi(0)$ and A vary linearly with the local stress σ .

$$\phi(0) = P_0 + P_1\sigma$$

$$A = A_0 + A_1\sigma.$$

The wall effect was dealt with by assuming that the stress decreased exponentially with distance down the column, expressed in terms of the weight of salt w :

$$\sigma(w) = \sigma(0) \exp(-w/c). \quad (5)$$

C is the weight of salt through which the stress falls by a factor of e . An expression for the observed void fraction is then derived by performing the integration:

$$\phi^0 = (1/W) \int_0^W \phi dw,$$

where W is the total weight of salt in the column. The resulting expression for the observed void fraction is:

$$\phi^0 = [P_0 + P_1\bar{\sigma}] + [(A_0 + A_1\bar{\sigma})/B] \ln [1 + Bt/\bar{z}(0)^3], \quad (6)$$

in which $\bar{\sigma}$ is given by:

$$\bar{\sigma} = (1/W) \int_0^W \sigma dw = \sigma(0) (C/W) [1 - \exp(-W/C)]. \quad (6a)$$

All the consolidation data in brine at room temperature from the 2.54-cm column have been fitted by least squares to Equation (6), to obtain the best set of the parameters A_0 , A_1 , B , and C . At the same time, the profiles of void fractions vs w from direct sectioning of consolidated columns were fitted by Equation (6), but with $\bar{\sigma}$ replaced by $\sigma(w)$ from Equation (5). The parameter values obtained were:

$$A_0 = (8.75 \pm 0.65) 10^{-9} \text{ (cm}^3 \text{ min}^{-1}\text{)}$$

$$A_1 = (3.94 \pm 0.33) 10^{-11} \text{ (cm}^3 \text{ min}^{-1} \text{ bar}^{-1}\text{)}$$

$$B = (3.21 \pm 0.24) 10^{-7} \text{ (cm}^3 \text{ min}^{-1}\text{)}$$

$$C = 212 \pm 22 \text{ (g)}.$$

In this fit the initial void fraction, given by parameters P_0 and P_1 in Equation (6), was allowed to assume the best value for each run, since it seemed clear from the data that the initial void fraction was not very reproducible. The average deviation calculated from the observed void fraction thus obtained was 0.0106.

The quantity B should be the rate of growth in volume of the average salt particle during consolidation. After a period of a week (10,000 min), the average size should be about 0.15 cm. This is much greater than any value that might be estimated from a section of the consolidated column shown in Figure 2. It may be appropriate, however, to take the effective particle size as that of an aggregate from which the brine has been excluded.

FIGURE 2. PHOTOMICROGRAPH OF A SECTION OF SALT, CONSOLIDATION RUN #130

Permeability of Single Crystal Salt Grain Boundaries to Brine

Measurements of the rate of change of permeation of brine between two single crystal surfaces of salt has continued. Axial crystal loadings were varied from 93.8 bars to as low as 4.4 bars and then explored in two sequences. The ratios of brine pressure to axial pressure have been kept to low values (between 0.11 and 0.01) so that the axial pressure is relatively undisturbed by the hydraulic pressure of the brine. Measurements to date have been made at $20.0\text{ C} \pm 0.05\text{ C}$.

All data, except the most current, have been added to the computer files to facilitate analysis. The definition of permeability (P_m) used for the present study is: P_m = grams of brine/minute/bar of brine pressure. The following expression has been assumed to represent the permeability as a function of time when the applied axial stress is constant:

$$\log P_m = \log P_1 - P_2 \log (P_3 + t). \quad (7)$$

Grams of brine passed (G) is the measured quantity and is related to the permeation by:

$$\frac{dG}{dt} = P_m \sigma, \quad (8)$$

where (σ) is the driving pressure of the brine in bars. When Equation (7) is substituted into Equation (8) and the integration performed, the following expression is obtained for the brine flow in the time interval t_1 to t_2 :

$$[G]_{t_1}^{t_2} = - \frac{\sigma P_1}{(P_2 - 1)} \left[\frac{1}{(P_3 + t) (P_2 - 1)} \right]_{t_1}^{t_2} \quad (9)$$

This expression was fitted by least squares to data over time periods in which the uniaxial stress applied to the crystals was constant. Some difficulty was experienced in refining the fit to Equation (9), and it was found more efficient to replace P_1 by

$$P_1 = P_5(P_3 + P_4)^{P_2}, \quad (10)$$

giving:

$$[G]_{t_1}^{t_2} = - \frac{\sigma P_5}{(P_2 - 1)} \left[\frac{(P_3 + P_4)^{P_2}}{(P_3 + t) (P_2 - 1)} \right]_{t_1}^{t_2} \quad (11)$$

Here P_3 and P_5 are adjusted along with P_2 . The parameter P_4 is an auxiliary time constant, a reference time centered in the time period of the measurements. The combination of adjustable parameters (P_2 , P_3 , P_5) shows less covariance than P_1 , P_2 , P_3 . The value of P_1 is obtained from Equation (10) after refinement.

In Figure 3, fitted results are shown graphically for the data set where the axial pressure was 46 bars and the brine pressure was 4.1 bars. Each bar indicates the integrated permeation over a time interval.

Data refinement by the above procedure is progressing. When enough cases have been refined, an analysis of the relationship between rates of permeation and the significant parameters of the system will be possible. A beginning has been made toward modifying the system to allow automated data collection.

Salt Core Brine Migration Experiment

Fabrication of the major components of the salt core holder was completed. The hydraulic ram was received and will be installed, together with equipment for maintaining constant pressurization over extended time periods. Equipment orders for temperature measurement and control, moisture measurement, and gas flow have been placed. It is planned to consolidate large particle size crystalline salt in brine in the initial experiments to shake down the equipment and to extend the consolidation experiments to 10-cm-diameter columns.

FIGURE 3. SINGLE SALT CRYSTAL PERMEATION FOR SATURATED BRINE AT 20 C AND 46 BARS AXIAL PRESSURE AFTER FITTING

REFERENCES

- Coble, R. L., 1961a. "Sintering Crystalline Solids. I. Intermediate and Final State Diffusion Models," *Journal of Applied Physics*, Vol. 32, No. 5, pp. 787-792.
- Coble, R. L., 1961b. "Sintering Crystalline Solids. II. Experimental Test of Diffusion Models in Powder Compacts," *Journal of Applied Physics*, Vol. 32, No. 5, pp. 793-799.

WBS 1.4.6

Project: Impact of Defense Waste on Repository Design

Principal Investigator: Los Alamos Technical Associates, Inc. (R. J. Kingsbury)

ONWI Project Manager: R. W. Brown

Objective

The principal objective of this project is to determine the impacts of receipt and emplacement of high-level and transuranic defense wastes on existing National Waste Terminal Storage geologic repository designs. Conceptual designs have been prepared or are in progress for underground nuclear waste repositories in salt and basalt geologic media. Analyses of these designs are to be performed with regards to the receipt and emplacement of nuclear wastes stored and generated at defense-related sites. Data and information on the physical, chemical, and nuclear characteristics of these wastes are to be collected and organized for use in the repository analyses. Impact analyses will include recommended conceptual design modifications required to mitigate the consequences of those significant impacts agreed to by ONWI. At ONWI's discretion, design criteria for subsequent repository design will be developed for each recommended design modification. A secondary objective is to determine the feasibility and costs of analyzing the impacts of receipt and emplacement of low-level defense wastes on the same repository designs.

Progress Reported Previously

This project was initiated on 12 August 1980. The collection of data and information on defense wastes resulted in the following general conclusions:

- Specific data are readily available regarding existing and projected quantities of high-level and transuranic wastes;
- All defense wastes will likely receive more treatment prior to delivery to a repository;
- Many options exist for the physical forms resulting from various treatment processes;
- A reference treatment process and a waste form have been recommended for high-level wastes;
- Options also exist for the canisterization of high-level and transuranic wastes.

Reviews of the collected defense waste documents were performed, and data and information were extracted regarding the following defense waste characteristics: physical form, existing quantities and locations, predicted and actual generation rates, chemical and nuclear compositions, treatment processes, and canisterization forms and configurations.

The allocation and distribution scenarios, specified by ONWI during the project initiation meeting, are based (among other constraints) on two repository designs—one in salt host rock and one in basalt host rock—and on two repository networks—one consisting of a single national repository and one consisting of regional repositories.

Activities During the Reporting Period

Objective

The work performed was to include: (1) completing the defense waste data collection, (2) developing scenarios for the distribution and allocation of defense wastes to national and regional repositories, (3) developing repository characterizations from existing conceptual designs, and (4) identifying the top-level repository impacts caused by the receipt and emplacement of defense waste.

Procedures

Defense waste data were collected during visits to the lead offices for the HLW Technology Program, the TRU Waste Management Program, and the Integrated Data Base Program. Review of these data has been performed, and the existing and future quantities, sources, locations, properties, characteristics, forms, and containers of defense wastes have been extracted and incorporated into a computerized data base. A data base management system allows manipulation of the defense waste data into lists based on key data parameters such as types, forms, sources, and containers. In addition, specific data can be retrieved singularly or in combination with other data and either displayed or printed. These data have been used by technical specialists during the identification and analyses of defense waste impacts on repositories.

The repository networks consist of the hypothetical siting of two repositories of differing design as a single national repository and as regional repositories. Four repository regions, western, southern, north central, and northeastern, have been identified. Impact analyses are being performed for each design at each hypothetical location with the following constraints:

- The designs considered are for the Conceptual Reference Repository Design (CRRD) and the National Waste Repository in Basalt (NWRB).
- All defense wastes are assumed to be stored in the national repository.
- Only the wastes from a given region are assumed to be stored in the repository for that region.
- The impact analyses will be performed on the repository designs consistent with their existing levels of detail and as they are specified by the respective design organizations.

Conceptual design descriptions and supporting cost data have been collected to the extent that such information and data are available. With regards to the CRRD, detailed drawings, specifications, and cost estimates are available for the repository areas translated directly from the NWTS-R1 and NWTS-R2 conceptual design reports. Areas derived from the Spent Fuel Receiving and Packaging Facility and originated by the CRRD design organization have less detail and supporting cost data. The conceptual design of the NWRB is currently in progress, and limited design detail is available in documented form. Also, no conceptual design cost estimates are planned for the NWRB by the design organization in sufficient time to be used during the impact analyses and design modification development.

One important achievement related to this work is the development and demonstration of the impact analysis methodology. A repository breakdown structure, for each repository, has been developed and incorporated in the impact analysis methodology along with the key defense waste data parameters. Generalized impact analyses have been performed using an interactive computer program developed for this purpose. The analysis program stores the identification of a waste on repository impact, the data and information resulting from the impact analysis, and other relevant information. These data, results, and comments are retrievable as lists based on waste data parameters, repository characteristics, and/or specific impacts.

Results

The reviews of the collected defense waste data and information have resulted in lists of parameters, data, and comments for differing wastes types, forms, and locations. Certain basic assumptions regarding the defense wastes are required. Typical of such assumptions are:

- The radionuclides of a given waste type will be assumed to be uniformly distributed in the product of a treatment process unless otherwise specified.
- All reasonable options for waste forms, treatment processes, and canisters will be considered during subsequent impact analyses.
- The reference HLW immobilization treatment process and canister are the primary options.
- TRU waste will be subjected to an immobilization treatment process that results in a 2:1 overall volume reduction.

The defense waste distribution and allocation scenarios have been based on the following repository networks specified by ONWI:

- *Network 1:* A single repository, location unspecified, to which all suitable defense wastes shall be delivered.
- *Network 2:* A single repository in each of four regions defined as western, north central, northeastern, and southern.

The regional repository network was further constrained by the requirement that all defense wastes generated, temporarily stored, and/or treated in a given region be shipped to the repository located in that region. Defense wastes suitable for repository terminal storage would not be transported across regional boundaries.

Lists of quantities and characteristics of defense wastes corresponding to the specified repository networks have been generated from the computer-managed data base. Data have been compiled from these lists to summarize the allocation and distribution of defense wastes to the national and regional repositories. These data pertain to existing stored and buried waste and future generated volumes. Data for contaminated soils and waste from commercial burial grounds have not been included.

Information on the repository network specifies: (1) the existing volume of stored waste, (2) the projected shipping schedule of defense waste canisters from an immobilization treatment facility, (3) the total number of canisters projected at year 2020, and (4) the total volume of immobilized waste to be emplaced by year 2020.

Based on the available conceptual design documents, repository breakdown structures have been developed for the CRRD and the NWRB. These breakdown structures have been incorporated into the LATA-developed, computer-aided, impact analysis methodology. The methodology has been used to systematically identify interactions between repository characteristics and key waste parameters.

Conclusions

The collection, review, and organization task of defense waste data and information is sufficiently complete and comprehensive for use during subsequent project tasks. Assumptions have been made where data, plans, and programs have not been fully defined.

The specified repository networks provide the bases upon which the repository impact analyses are being performed. When completed, the results will have a parametric nature that will be applicable to a broad scope of actual, defense waste cases.

Identification of defense waste impacts on repositories using the analysis methodology allows easily retrieved results, traceable throughout the repository characterization.

Future Activities

The following project activities are planned:

- The identification of impacts on CRRDs caused by the receipt and emplacement of defense waste will be completed.
- Descriptions of these impacts will be developed.
- A methodology will be developed to rank all identified impacts in terms of their importance to repository performance objectives.

WBS 1.4.8

Project: Material Properties Testing and Analysis

Principal Investigator: RE/SPEC, Inc.

ONWI Project Manager: M. R. Wigley

Thermo/Mechanical/Hydrological Analyses
(G. D. Callahan and A. F. Fossum)

Objective

The objective of this program is to assess the thermomechanical behavior and constraints of repository situations in representative geologic formations by numerical analysis procedures (involving the canister scale, room scale, and repository scale regions) and by data from appropriate laboratory testing, laboratory bench scale testing, and in situ experiments. The resulting work efforts address and attempt to answer particular problems relative to the Rock Mechanics Program of the (NWTs) program.

Progress Reported Previously

This program was initiated on 1 October 1978.

During the FY 80 rock mechanics numerical modeling analyses, attention was focused on canister scale, room scale, and repository scale regions of a repository system and the relative effects of each region on the overall geologic containment of radiogenic material.

At the canister scale, it was concluded that the thermomechanical behavior of the region near the canister could be assessed adequately by addressing the coupled influence of temperature, stress, and brine migration.

At the room scale, a preliminary investigation was completed of structural behavior peculiar to the three-dimensional geometry of entry-entry intersections and to inhomogeneous initial stress fields. The corner of the pillar (at the entry-entry intersection) was pointed out as a potential problem area not identifiable with two-dimensional modeling. Moreover, it was concluded that inhomogeneous initial in situ stress fields could have a significant impact on the nonlinear deformation response around a room and should therefore be considered in subsequent analyses.

A repository scale analysis was initiated in FY 80. Repository scale models generally neglect the room-and-pillar detail prevalent in the room-scale models in favor of including the geological detail between the repository and the ground surface. This analysis is continuing in FY 81.

Activities During the Reporting Period

Technical Assistance on Special Repository Problems

The RE/SPEC finite element code series SPECTROM is being compared with the SAI explicit finite difference code STEALTH for a room-scale benchmark problem. The benchmark problem involves a thermo-viscoelastic analysis of a repository in salt at a depth of 610 meters. During this quarter, these analyses were completed. Calculations involved the implementation of a strain-hardening technique for evaluating the accumulation of creep-strain. Previous calculations were made with the time-hardening technique. Figures 1 and 2 illustrate the vertical displacements at the centerline of both the roof and floor for the two techniques. Displacements are greater for the strain-hardening case.

Code Development and Verification

The SPECTROM series of computer programs undergoes continuous development and verification as required to provide state-of-the-art analysis capability for rock mechanics problems unique to the ONWI program. During this quarter the special-purpose heat transfer code, SPECTROM-41, was updated to permit explicit definition of transient temperature boundary conditions. Previously, approximations of transient temperature boundaries were achieved implicitly by updating the constant temperature boundaries at discrete time intervals to define a quasi-steady-state representation of the transient temperature boundaries. The new procedure requires one-fifth the number of time steps required previously to maintain the same solution accuracy. Work was initiated to develop a code capable of modeling the combined conductive-convective heat transfer process to permit an analysis of blast cooling of a repository prior to retrieval. Code modifications were proposed to eliminate numerical instabilities encountered during initial program runs.

Development of SPECTROM-321, a special-purpose, three-dimensional, thermo-viscoelastic finite element program continued during this quarter. Satisfactory 3-D results were obtained for comparative 2-D situations involving plane strain and for problems with known analytical solutions. This code will be used for the room-scale investigations.

Documentation of SPECTROM-41 continued with the formulation of example problems. The user's manual is approximately half completed.

Future Activities

During the next quarter, the nonlinear effects in three-dimensional room and pillar geometries in salt will be investigated. Particular attention will be given to the corner of the pillar (at the entry-entry intersection), pointed out as a potential problem area in FY 80. The analysis will also include a two-dimensional comparison to determine regions of similarity and the emplacement of heat-generating waste to induce thermal stresses, as well as those created by the initial excavation of the disposal room. In summary, this task will include thermoelastic, thermoelastic-plastic, and thermo-viscoelastic analyses of a room and pillar model in a salt environment. These analyses should be nearing completion by the end of the next quarter.

FIGURE 1. TIME HISTORY OF THERMO-VISCOELASTIC VERTICAL DISPLACEMENTS AT THE CENTERLINE OF BOTH THE ROOF AND FLOOR (TIME HARDENING)

FIGURE 2. TIME HISTORY OF THERMO-VISCOELASTIC VERTICAL DISPLACEMENTS AT THE CENTERLINE OF BOTH THE ROOF AND FLOOR (STRAIN HARDENING)

Also during the next quarter, modifications should be completed for the finite element code capable of modeling combined conductive-convective heat transfer so that an analysis of blast cooling can begin.

Laboratory Studies

(P. E. Senseny)

Objective

The objective of this investigation is to study in the laboratory the constitutive behavior of geologic materials and to formulate constitutive laws that model the thermomechanical response of these materials under conditions expected in a nuclear waste repository.

This program provides constitutive laws that can be used by modelers to predict the stresses and deformations in a nuclear waste repository and to assess the stability of that repository.

Progress Reported Previously

Previous work involved study of the creep of rock salt. In particular, specimens of dome salt from Avery Island, Louisiana, were allowed to creep in axisymmetric compression at temperatures up to 200 C. Transient creep data were fit to a power law:

$$\epsilon = A\sigma^m t^n T^p$$

where ϵ is axial creep strain, σ is axial stress difference, t is time since the start of creep, T is absolute temperature, and A , m , n , and p are constants. Two different specimen sizes were employed in this study: 50 mm and 100 mm in diameter. Analysis of the creep data showed that transient creep of the smaller specimens is more sensitive to stress, time, and temperature than is that of the large specimens.

Activities During the Reporting Period

During this period the problem of large oscillations in the confining pressure was finally solved. In the laboratory testing machines, the confining pressure was controlled by a mechanical differential pressure gage that compared the pressure in the pressure vessel with that in a closed tube, the reference pressure. When the two pressures were different, the differential pressure gage deflected and triggered a pump to add or remove oil from the pressure vessel. The pump removed oil if the pressure in the vessel was greater than the reference pressure and added oil if it was less than the reference pressure. While this system is very simple, it did not work well because the reference pressure changed with temperature fluctuations in the laboratory.

The mechanical differential pressure gage was replaced with an electronic module that compares the output voltage of a pressure transducer with a reference voltage. If the output

voltage is greater than the reference voltage, the electronic module commands the pump to remove oil from the pressure vessel. If the output voltage is less, then the module commands the pump to add oil to the pressure vessel.

Figure 3 plots confining pressure as a function of time for an experiment during which the control unit was changed. In the first part of the experiment, the confining pressure was controlled by the mechanical differential pressure transducer. The oscillations in confining pressure are quite large (~ 100 psi). In the last part of the experiment, the electronic module controlled the confining pressure and the oscillations are reduced to only several psi.

These electronic modules have been installed on all six of the creep machines at RE/SPEC.

Future Activities

Future activities include performing tests to study the influence on creep of stress and temperature history.

Avery Island Data Analyses (R. A. Wagner)

Objective

The objective of this task is to perform analytical and numerical evaluations of the field experiments performed at Avery Island, Louisiana. Present and future field experiments may be categorized as follows:

1. Heater
2. Brine migration
3. Accelerated borehole closure
4. Extended gas permeability
5. Fracture healing
6. Heated block
7. Rock mechanics measurements.

The analyses or simulation of these experiments will help assess the in situ thermomechanical properties and evaluate the modeling techniques of situations related to the isolation of radioactive waste.

Progress Reported Previously

Simulation of two field experiments (heater and accelerated borehole closure) have been in progress prior to this reporting period and are discussed below.

Three heater tests designated as Sites A, B, and C were in operation by June, 1978. The numerical simulation of Site A was initiated at approximately the same time. Presently, the numerical analyses of the heater tests involve the simulation of Site A and the comparison of

FIGURE 3. PLOT OF CONFINING PRESSURE VERSUS TIME SHOWING IMPROVEMENT OBTAINED BY REPLACING MECHANICAL DIFFERENTIAL PRESSURE GAGE WITH ELECTRONIC MODULE

these results to the field measurements. In general, the agreement between the numerically predicted response and field measurements is fairly good.

The numerical simulation of the accelerated borehole closure field experiment was initiated in June, 1980. This particular numerical simulation was performed prior to the operation of the actual field experiment. Because of operational difficulties in the field experiment, the loading-heating sequence was modified such that an additional numerical simulation may be necessary to encompass all aspects of the present tests. Preparations are now being made to repeat the accelerated borehole closure tests using the actual loading-heating sequence.

Activities During the Reporting Period

Activity has been limited to the composition of the draft report concerning the numerical simulation of the accelerated borehole field experiment.

Future Activities

The future activities of the Avery Island studies include the ongoing numerical simulation of the heater experiments. This work will involve the completion of the report concerning the Site A simulation and the comparison of these predicted results with the actual field measurements. An additional investigation may be initiated during this next quarter with regards to consolidation of crushed salt. This consolidation analysis will be interrelated with the accelerated borehole closure analysis. Basically, the variation between these two analyses will involve the addition of crushed salt within the concentric borehole.

WBS 1.4.8

Project: Convective Heat Transfer in Room and Pillar Geometries

Principal Investigator: Ohio State University (R. Christensen)

ONWI Project Manager: G. E. Raines

Objective

The objective is to determine the local and average convective heat transfer coefficients on the surfaces of a repository mine drift, and to measure the local temperature distribution in the drift. These heat transfer coefficients are needed to assess the ventilation and cooling requirements in the repository during waste canister emplacement and retrieval. Ventilation is required to ensure worker safety and may be required to limit the temperatures in both the repository and the waste package.

Progress Reported Previously

A basic research effort was started in July, 1979, to determine generalized heat transfer correlations for all interior surfaces of a room, using discrete thermal sources on the floor.

Typical room designs were determined via a literature survey and in consultation with other waste isolation programs (e.g., WIPP and Avery Island). Once a range of design parameters had been determined, an experimental design study developed into a dynamically scaled model for laboratory experiments. This model was scaled to provide hydrodynamic and thermodynamic similarity. The dimensionless parameters that come into play are the Reynolds number, the Prandtl number, room aspect ratios (i.e., length to width, height to width), thermal loading schedule, total thermal load, and surface roughness. The laboratory model was two-dimensional and had a strip heater in the floor (to represent the heat generating waste) and insulated side walls.

Heat transfer coefficients were obtained in the fully developed turbulent region of the channel over a range of Reynolds numbers from 1,600 to 88,700 for two different canister spacings, two heater powers, and various waste loading sequences. These data agree favorably with correlations for mixed convection over the entire range of Reynolds numbers. They also agree with forced convection correlations at the highest Reynolds numbers ($Re > 85,000$).

Activities During the Reporting Period

During the quarter, significant progress was made in three areas: (1) continued accumulation of data for the smooth-walled case, (2) development of a computer program for data analysis, and (3) linear regression analysis for the development of a preliminary correlation for asymptotic Nusselt numbers in the fully developed turbulent flow regime.

Data have been accumulated for varying Reynolds numbers, heater power, heater spacing, and heated length. The ranges of these variables are presented in Table 1, and Table 2 presents the heating sequences used.

**TABLE 1. RANGE OF OPERATING CONDITIONS FOR EXPERIMENTAL
REPOSITORY DRIFT MODEL**

Parameter	Range of Values
Approximate Reynolds Numbers	9,000, 16,000, 32,000, 63,000, 89,000
Heater Power	0.5 W, 5.0 W
Heater Spacing	2.54 cm (adjacent heating) ^(a) 5.08 cm (discrete heating) ^(b)
Heated Lengths	Heating Sequences Q through Z

(a) Every heater is activated.

(b) Every other heater is activated.

TABLE 2. HEATING SEQUENCES

Heating Sequence	Heaters Activated	Heating Sequence	Heaters Activated
A	1	N	1-14
B	1-2	O	1-15
C	1-3	P	1-16
D	1-4	Q	1-24
E	1-5	R	1-32
F	1-6	S	1-40
G	1-7	T	1-48
H	1-8	U	1-64
I	1-9	V	1-80
J	1-10	W	1-96
K	1-11	X	1-112
L	1-12	Y	1-144
M	1-13	Z	1-180

For a heater power of 5 W per heater, the lowest Reynolds number (9,000) was not run because of temperature limitations on the channel materials. Heating Sequences A through P have been run previously. These data, in addition to those gathered previously, will provide correlations for an average Nusselt number as a function of Reynolds number, heater spacing, and heated length. These correlations will allow more accurate calculations of heat removal rates in a ventilated repository drift.

A computer code was developed to calculate the following for each run: air temperature at the channel center line (T_{CL}), nondimensional temperature $(T_W - T_{CL})/T_{CL}$, heat transfer coefficient (h), Nusselt number ($h D_h/k_A$), and Grashof number ($g\beta L^3 \Delta T/\nu^2$) as a function of nondimensional distance (x/D_h). In the asymptotic region, the heat transfer coefficients,

Nusselt numbers, and Grashof numbers were averaged to find an asymptotic heat transfer coefficient (h_∞), Nusselt number (Nu_∞), and Grashof number (Gr_∞). Values for the thermal conductivity of the air (k_A) and the viscosity of the air (ν) were obtained from a best fit equation provided by the *National Bureau of Standards Circular 564*. These properties were evaluated at the average film temperature $(T_{CL} + T_W)/2.0$. The Reynolds number for each run was calculated using a special method by Spink (1967) and the principle of conservation of mass. This computer program allows systematic reduction of data taken over the last two calendar quarters.

Of all the data accumulated so far, asymptotic Nusselt numbers have been obtained for data in the fully developed turbulent flow regime by use of the aforementioned program. These Nusselt numbers have been plotted versus Reynolds numbers in Figure 1 and are compared with various forced convection and mixed convection correlations in ONWI-148 (1980). In addition, these data have been fit via linear regression techniques by the equation

$$Nu_\infty = 4.17 Gr^{0.0137} Re^{0.300}$$

This correlation is also plotted in Figure 1. It should be emphasized that this equation is a preliminary fit of the data. As more data are analyzed, the value of the constants may change slightly. However, the general form of the equation and the order of magnitude of the constants should not change.

FIGURE 1. ASYMPTOTIC NUSSELT NUMBERS AND COMPARISON WITH FORCED CONVECTION AND MIXED CONVECTION CORRELATIONS

Future Activities

During the next quarter data analysis will continue. This will allow formulation of correlations for average Nusselt number (Nu_{AVE}) that will include dependence on the heated length. The channel will be modified to allow for runs with range surface roughnesses.

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Spink, L. K., 1967. *Principles and Practices of Flow Meter Engineering*, ninth edition, Foxboro Company, Foxboro, MA.

WBS 1.4.8**Project:** Convective Heat Transfer Within Spent Fuel Canisters**Principal Investigator:** Ohio State University (F. A. Kulacki)**ONWI Project Manager:** G. E. Raines**Objective**

The objective is to provide basic engineering data and correlations for heat transfer from rod bundles typical of spent fuel subassemblies in enclosed canisters. The heat transfer coefficients are to be obtained from laboratory measurements and will be corrected for the effects of thermal radiation. Bundles containing a 5 x 5 rod array and smaller arrays will be considered to provide a basis for a generalized heat transfer correlation. Constant temperature and mixed type boundary conditions on the canister wall will be used in separate phases of the study.

Progress Reported Previously

Previous work on this task has included the design, construction, and testing of a vertical circular cylinder (referred to as the "small apparatus") for studies of free convection from a single rod (annulus case) and rod bundles containing four, five, and nine rods. In all cases, the rods are electrically heated with specific power generation scaled to that expected in an actual fuel rod bundle—PWR fuel with 33,000 megawatt day (MWD) burnup, 5 years old. At the end of FY 80, heat transfer results for the annulus case were obtained for the laminar (conduction), transition, and turbulent regimes of free convection. In the laminar regime, the heat transfer coefficients were in general agreement with the numerical predictions of Thomas and de Vahl Davis (1970), who considered laminar convection in an annulus with two constant temperature walls. In the present experiments, only the outer wall was maintained at constant temperature, while the inner wall assumed a temperature distribution with a peak, or maximum, above the mid-plane of the cylinder. This behavior of the rod temperature is similar to that predicted by McCann via numerical techniques for rod-bundle heat transfer and that found in measurements of actual rod bundles in full-scale canisters.

Activities During the Reporting Period**Heat Transfer Measurements**

The experiments with the single rod, that is, the annulus case, were completed, and two correlations have been developed for average heat transfer coefficients, or Nusselt numbers, as a function of Rayleigh number based on gap width and average temperature difference between the walls of the annulus. The experimental data span the laminar, transition, and turbulent, or boundary layer regimes of convection. For the laminar (sometimes referred to as the conduction regime), the correlation is

$$Nu = 1.40 Ra^{0.077} \quad (1)$$

for $10^3 \leq Ra \leq 6 \times 10^3$. Heat transfer data for the transition and boundary layer regimes could be combined into one correlation, since the measurements did not permit a sharp distinction to be made between the two regimes of convection. The correlation is

$$Nu = 0.163 Ra^{0.322} \quad (2)$$

for $6 \times 10^3 \leq Ra \leq 2.4 \times 10^6$.

A comparison of the present results with results in the open literature is not possible, since no similar study exists. However, it is possible to modify the results of Sheriff (1966), who also experimentally studied the vertical annulus with thermal boundary conditions similar to those of the present work.

The result reported by Sheriff (1966) of interest to the present work is a correlation for Nusselt number as a function of Rayleigh number and the aspect ratio, H ($H = L/(R_o - R_i)$), where L = height of the cylinder and $R_o - R_i$ = the gap width. His correlation is

$$Nu = 0.25 Ra^{0.30} H^{-0.25} \quad (3)$$

for $Ra \geq 10^5$.

Sheriff's experimental data included three radius ratios, $K = 1.23, 1.10$, and 1.03 , where $K = R_o/R_i$, with the working fluid being carbon dioxide ($Pr = 0.76$). The experiments were conducted under pressure to extend the range of Rayleigh numbers.

In order to modify Sheriff's correlation for a comparison with the results of the present work, the correlation reported by Thomas and de Vahl Davis (1970) has been used. For the boundary layer regime, they reported that the Nusselt number dependence on the radius ratio was $K^{0.442}$ and the Nusselt number dependence on the Prandtl number was $Pr^{0.006}$. Assuming that this same dependence holds for boundary conditions of constant heat flux, as well as constant temperature, one can modify Sheriff's results to the following:

$$Nu = 0.239 K^{0.442} Pr^{0.006} Ra^{0.30} H^{-0.25} \quad (4)$$

where the leading factors in Equation (4), $0.239 K^{0.442} Pr^{0.006}$, have been formally set equal to 0.25 for $Pr = 0.76$ and an average radius ratio over the range of ratios used by Sheriff.

The graphical comparison of the present results with the modified results of Sheriff is given in Figure 1. For the region of the validity that applies to both results, $10^5 \leq Ra \leq 2.4 \times 10^6$, the two results are in good agreement.

Design of Large Apparatus

The large experimental apparatus has been designed and is presently under construction.

FIGURE 1. COMPARISON OF THE PRESENT RESULTS WITH EQUATION (4)

Future Activities

A series of experiments with a 3 x 3 rod bundle square array will be conducted to establish the effect of radius ratio on the heat transfer from the rod bundle to the cylinder. In particular, the mean heat transfer coefficient for the bundle will be defined, and ways will be sought to adequately represent the data in dimensionless form.

Experiments will be conducted first with air and then with helium as the test fluid. The experimental apparatus already in operation will be used for these tests.

Construction and instrumentation of the large apparatus will continue throughout the next quarter. It is estimated that it will be ready for initial test runs during FY 81.

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WBS 1.4.8

Project: The Effects of Far-Field Inhomogeneities on Surface Temperature and Heat Transfer From a Repository

Principal Investigator: Ohio State University (A. T. Conlisk)

ONWI Project Manager: G. E. Raines

Objective

The objective is to calculate approximate temperature profiles within geologic media that are subject to a spatially finite, time-dependent heat source. Property variations are to be incorporated into the model, and various source (repository) geometries will be considered. The corresponding linear (constant property) solutions will be compared with the nonlinear (variable property) solutions. Where possible, analytical closed-form solutions to the fundamental partial differential equation will be given; in the nonlinear problem, such a solution does not appear possible and the temperature distribution within the geologic medium will be computed numerically. A few of these solutions will be compared with the experimental results from a laboratory scale model of the repository and its subsurface surroundings, as well as with the relevant existing numerical calculations.

Progress Reported Previously

The heat balance integral method has been successfully applied to a semi-infinite domain containing planar discontinuities, with the heat source (repository) being represented by either a planar source or a finite line source. The method has yielded the time history of the temperature distribution within the geologic medium with a minimum of complexity and numerical computation. However, the studies to date do not appear to take into account the nonlinear nature of the problem, although mechanisms for generalizing the method to this case appear to exist.

Consequently, work has begun in an attempt to rigorously evaluate the effect of a temperature-dependent conductivity. To do this, the problem has been formulated as a nonlinear, parabolic, partial differential equation to be solved by advanced numerical techniques. The corresponding linear problem can be solved by employing Laplace transforms. The description of the problem and the results of comparing the linear with the nonlinear solution follow.

Activities During the Reporting Period

The problem of determining the temperature distribution within a geologic medium which is driven by a heat source due to contained nuclear waste has been investigated analytically and numerically. The analytical dimensionless formulation is summarized in Figure 1; as depicted in this figure, the solution for the temperature distribution requires solution of a nonlinear ($\epsilon \neq 0$ in Figure 1) partial differential equation. It has been of interest in this phase of the study to compare the temperature distribution for the constant media thermal conductivity case (linear, $\epsilon = 0$) with that for the nonlinear ($\epsilon \neq 0$) case.

FIGURE 1. DEFINITION OF THE PROBLEM OF DETERMINING THE TEMPERATURE DISTRIBUTION WITHIN GEOLOGIC MEDIA DRIVEN BY A HEAT SOURCE (NUCLEAR WASTE REPOSITORY) AT $y = L$

The surface $y = 0$ is the surface of the earth, q is the heat generation rate, and $T_G(y)$ is the geothermal temperature distribution in the absence of the repository.

In this study, the variation of the thermal conductivity of the geologic media has been modeled as quoted previously according to:

$$k = \frac{1}{a_0 + a_1 T^*} \quad (1)$$

where a_0 and a_1 are best fit constants. Nondimensionalizing the problem indicates the definition of the parameter:

$$\epsilon = \frac{a_1}{a_0} T_\infty; \quad (2)$$

the temperature has been scaled on T_∞ as

$$T^* = \frac{T - T_\infty}{T_\infty} \quad (3)$$

The limit $\varepsilon \rightarrow 0$ yields the approximation of constant thermal conductivity within the geologic formation.

Solutions were computed for the case of buried CANDU fuel, where the generation or decay rate of the fuel has been modeled as in Beyerlein and Claiborne (1980); the generation rate has been normalized to unity at time $t = 0$ in all calculations. For $\varepsilon = 0$, an analytical solution can be computed by Laplace transform techniques; this result is given and discussed below. The solution for $\varepsilon \neq 0$ must be computed numerically; in this study, the Crank-Nicholson (Smith, 1965) technique was employed to integrate the partial differential equation forward in time. Such a method ensures quick and accurate solutions to the difference equation analog of the problem in Figure 1. Numerical solutions were obtained for granite, where $a_0 = 0.3514$ and $a_1 = 0.0003795$; the calculations were made for $L = 1,000$ meters and $T_\infty = 27$ C. For granite, these parameters correspond to $\varepsilon = 0.32$; the geothermal gradient at the surface of the earth has been assumed to be ~ 3 C/100 meters. The analytical series solution discussed in the Attachment is slowly converging for small times and, consequently, there appears to be no loss of clarity in computing numbers for the linear solution numerically as well. To check for accuracy, numerical solutions were computed for a dimensionless time step of 0.0001, which corresponds to a dimensional time step of 2.69 years; the time step was reduced, and the two solutions were checked. At least three-figure accuracy was found in the solutions, and greater accuracy was obtained at larger times. The 21 spatial mesh points across the layer were found to yield three-figure accuracy as well.

The results of the numerical calculations are presented in Figure 2, where the temperature of the source is computed as a function of time. This figure may be interpreted as an evaluation of the validity of the assumption of constant thermal conductivity. Previous studies (Beyerlein and Claiborne, 1980; Amar and Rickertsen, 1980) have generally assumed constant conductivity; in Figure 2, these linear solutions correspond to the solid curve. Note that for $\varepsilon = 0$ in Figure 2, the shape of the curve closely approximates that reported in Beyerlein and Claiborne (1980). In the early stages of calculation (~ 100 years), the nonlinear problem gives roughly a 6 percent rise in dimensionless temperature over the linear problem; however after roughly 10,000 years, the same figure is up to about 25 percent. Moreover, although it is not evident on this log plot, the second peak of the nonlinear problem occurs at a much later time ($\sim 18,000$ years) than the linear solution ($\sim 10,000$ years). Apparently, Figure 2 indicates that the varying conductivity of the geologic medium may have significant effects upon the temperature distribution within the geologic medium. Consequently, it appears that further work in this program may require analysis and solution of the temperature-varying conductivity problems.

Attachment: The Linear Solution

For $\varepsilon = 0$, the solution to the problem depicted in Figure 1 is given by:

$$T = \bar{C}_y + q_0 \sum_{i=1}^N C_i e^{-\varepsilon_i t} \frac{\sin \sqrt{\varepsilon_i} y}{\sqrt{\varepsilon_i} \cos \frac{\sqrt{\varepsilon_i} L}{2}} - \frac{16}{\pi^2} \sum_{n=1}^{\infty} \left\{ q_0 \sum_{i=1}^N \frac{C_i}{\frac{4\varepsilon_i}{\pi^2} - (2n-1)^2} \right\} (-1)^n \sin \left(\frac{2n-1}{2} \right) \pi y e^{-(2n-1)^2 \pi^2 t/4}, \quad (4)$$

FIGURE 2. COMPUTED SOLUTIONS FOR THE TEMPERATURE AT THE NUCLEAR WASTE REPOSITORY AS A FUNCTION OF TIME FOR A SINGLE LAYER OF GRANITE

The dashed curve denotes the nonlinear solution for $\epsilon = 0.32$; the solid curve corresponds to the linear solution which is given in Equation 4.

where q_0 is a scale factor. The dimensionless heat flux at the repository generated by the CANDU waste is given by

$$q = q_0 \sum_{i=1}^N C_i e^{-\epsilon_i t}, \quad (5)$$

where C_i and ϵ_i are dimensionless constants; g_0 has been chosen so that $g(0) = 1$. The C_i correspond to those given in Beyerlein and Claiborne (1980), and $\epsilon_i = \lambda_i \tau$, where λ_i are those also given in Beyerlein and Claiborne (1980); further, $\tau = \rho c a_0 L^2$, which is the time scale of the granite geologic formation ($\sim 26,000$ years for chosen parameters). Also in (4), $\bar{G}$ is the geothermal flux which has been assumed to be 3 C/100 meters (Kappelmeyer and Haenel, 1972).

Equation 4 is strictly valid only in the limit $\epsilon \rightarrow 0$ and is only approximate for $\epsilon \neq 0$; for $\epsilon \geq 0.1$, Equation 4 is no longer accurate. In the literature, it has usually been assumed (in the one-dimensional case, as well as in two and three-dimensional problems) that solutions similar to Equation 4 represent, in some sense, the temperature distribution within the geologic media. However, even for the case of granite, assuming the variation of thermal conductivity as in Equation 1, only for small values of T_∞ will ϵ be small. Consequently, linear solutions similar to Equation 4 appear to have limited validity under actual geologic and repository conditions.

Future Activities

During the next few months, nonlinear calculations will be made for a two-layer model; the corresponding dimensional problem for both the one and two layer models as formulated will be solved in an attempt to compare the results with the earlier work of Beyerlein and Claiborne (1980) and Kappelmeyer and Haenel (1972). Concurrently, asymptotic methods will be formulated in an attempt to obtain analytic closed-form solutions for small but nonzero ϵ . Moreover, the two-dimensional problem will be formulated, whereby the repository is modeled as a finite heat source of a specific shape. Experimental measurements will be taken and the results compared with those from the analytical model.

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WBS 1.4.8

Project: Development of ROCMAS (Thermo-Mechanical-Hydrological) Code

Principal Investigator: Lawrence Berkeley Laboratory (C. F. Tsang and J.S.Y. Wang)

ONWI Project Manager: J. O. Duguid

Objective

The objectives of this project are to develop a model simulating thermo-hydro-mechanical flow in fractured rock masses and to monitor the state of the art in modeling of fractured systems. This modeling capability is needed to predict the effects of waste heat and to assess the movements of ground water and possible radionuclides between the repository and the biosphere.

Progress Reported Previously

FY 80 work covered two main areas. Initially, efforts focused on assessing the current state of the art in modeling. Later efforts concentrated on working toward a fully coupled thermo-mechanical-hydrological model based on one of the most promising existing models as determined in the state-of-the-art assessment, the LBL model known as ROCMAS.

The ROCMAS code, used to complete the coupled stress and flow analysis, was made available on file at LBL's Computer Center.

Activities During the Reporting Period

The development of the thermo-mechanical-hydrological fractured rock code ROCMAS continued. The main activities were:

1. An application to the case of fluid injection into a deep-lying horizontal aquifer, under conditions of both constant head and constant flow, was solved using a preliminary version of the ROCMAS model.
2. A steady-state algorithm for coupled stress and flow analysis was developed.
3. The testing of the thermally induced, coupled hydro-mechanical code began.

The draft paper, *Status of Modeling Flow and Transport in Fractured Rock Masses as Related to Radioactive Waste Isolation* outlines the state of the art of modeling in this way:

1. Introduction

- General discussion of the use of models
- Description of discrete, double-porosity, and porous media models
- Layout, somewhat detailed, of what modeling capabilities are needed in waste management and why they are important
- Discussion of model uses in waste management, hydrology, geothermal energy, and the petroleum industry.

2. Governing Equations

- Fluid flow; discrete, porous medium, and double porosity
- Rock deformation
- Heat transfer
- Solute transport.

3. Numerical Methods and Computer Codes

- Brief discussion of advantages and disadvantages of finite difference, integrated finite difference, and finite element
- Problems that arise by choice of time stepping
- Matrix solvers and efficiencies of various techniques.

4. Validations and Applications

- Analytical solutions
- Laboratory and field testing
- History matching.

5. Discussion

- General discussion of when and at what stage of repository development advanced (coupled) models are needed.
- Research needed to develop these models; flow, transport (mass and heat), thermo-mechanical, and thermo-mechanical-hydrological.

6. Conclusions

- Where is the emphasis needed, and what are the priorities to reach the goals set out in the introduction?

Future Activities

The testing of the thermally induced, coupled hydro-mechanical code will continue, and the paper outlined above will be finalized.

WBS 1.4.8

Project: Analysis of Thermal/Mechanical/Hydrological Response of Rock

Principal Investigator: Science Applications, Inc. (R. Hofmann)

ONWI Project Manager: M. R. Wigley

Objective

The objectives of this study are: (1) to further develop and verify special-purpose versions of the computer code system STEALTH, which will be used to aid in the design and licensing of a nuclear waste repository; (2) to document this development, verification, and use of the STEALTH computer code system as it applies to geologic isolation of nuclear wastes; and (3) to perform generic design studies that will form the design and licensing data base for site-specific analyses.

Progress Reported Previously

The project was started 1 October 1980; therefore, this is the first quarterly technical progress report.

Activities During the Reporting Period

The purpose of this quarter's work in relation to the program as a whole has been to establish the base for the various simulations that must be performed in the months to come. The steps taken to reach this end included: (1) rerunning previous simulations on a PRIME 750 computer to verify hardware independence of the STEALTH codes; (2) writing a Quality Assurance (QA) manual which would govern ONWI-specific STEALTH developments; and (3) creating a base of STEALTH files for use in future calculations.

So far, the steps to achieve these goals have all been accomplished with minimal difficulty. Old Project Salt Vault (PSV) and Stripa runs have been successfully rerun, using current STEALTH files, version 3-2i. Also, ONWI-specific STEALTH updates have been converted to ATLAS.

Problems

So far, only one technical problem has occurred. After reviewing the South Dakota School of Mines uniaxial experiment, it was determined that it is an inappropriate test simulation. As a result, it will be replaced by a parametric set of calculations suggested by ONWI.

Future Activities

Three main milestones are anticipated in the quarter starting in January, 1981:

1. Complete a new three-dimensional PSV simulation;
2. Recalculate the Stripa runs using updated CAVS;
3. Begin the parametric calculations.

SUMMARY

1.5 REGULATORY AND INSTITUTIONAL

Work conducted under this program element includes the development of licensing strategies, criteria, and guides necessary to identify the required information for, and the preparation of, licensing documents. The ultimate objective is the granting of an operating license for a geologic repository by the Nuclear Regulatory Commission.

Work in the first quarter of FY 81 included NRC interactions, a meeting of the Licensing Interface Coordination Group (ICG), development of regulatory baseline documentation, and efforts to implement the licensing topical report plan.

An informal meeting was held on 14 November, 1980, in Columbus, Ohio, with representatives of the NRC, DOE, and ONWI. The purpose of the meeting was to discuss the NWTS approach to model development and the relative scope of the NWTS and NRC model development programs. The meeting resulted in a better understanding by both groups of the model development efforts presently under way. Licensing Program Office (LPO) personnel attended an Advisory Committee on Reactor Safeguards (ACRS) subcommittee meeting in Washington DC on 13 and 14 November. The purpose of the ACRS meeting was to discuss the Confidence Rulemaking Position Statements and Cross Statements. A consultant to the LPO attended an NRC commissioner briefing on proposed 10 CFR Part 60 on 22 December, 1980. The purpose of that briefing was to discuss the adoption of the proposed 10 CFR Part 60 Administrative. The commissioners remanded the rule back to the staff for further modification.

The second Licensing-ICG meeting was held in Albuquerque, New Mexico, on 11 and 12 November, 1980. The L-ICG approved the action plan, including planned document review schedules and identified issues, for that ICG, was briefed on the issues set forth in Part 2 of the Licensing Plan, and received the draft Licensing Plan for review and comment by January, 1981. Subsequent to that meeting, the Licensing Topical Report (LTR) Plan was distributed to the L-ICG for comment, with a request for project feedback on LTR scheduling priorities and participation in LTR development by project personnel.

The draft NWTS Licensing Plan and the draft Licensing Topical Report Plan were transmitted to NPO for review and comment. Both plans had been reviewed by ONWI management prior to transmittal, and the resultant comments were incorporated. The revised Environmental Report outline (ONWI-61) was transmitted to NPO, the NWTS projects, and WIPP for review and comment. The revised draft includes extensive revisions that were made to reflect previous NWTS comments and informal NRC format and content guidance. Workshops were held to develop the Q-List and the Safety Analysis Report outline and matrix in Columbus on 27 to 29 October, 1980. The workshops included ONWI and ONWI subcontractors. Workshops were also held to review and evaluate Preliminary Information Report (PIR) comments, including those received from NPO in December. The PIR development was on schedule through the first quarter of FY 81. However, the late and unexpected receipt of voluminous DOE-HQ comments caused a major perturbation in the effort to complete the PIR on schedule.

Efforts continued to develop the Safety Assessment Plan. A first draft of the Safety Assessment Plan was prepared and reviewed internally, and efforts commenced to prepare a second draft intended for ONWI Management review.

Action plans were developed for the Licensing Topical Reports on seismicity, volcanism, brine migration, quality assurance, performance modeling relative to site selection, and human interference. The Human Interference Task Force meeting was held in Chicago on 5 and 6 November. At that time the outline for the Topical Report was completed, the logic diagram for the report was developed, and efforts were initiated to identify expert consultants required to support portions of the logic diagram. A meeting was also held with the Associate Dean of Engineering at MIT and several of his colleagues to discuss the Human Interference LTR. The MIT participants in the meeting were supportive of the approach used and expressed interest in reviewing the LTR after it was drafted.

The second annual information meeting of the NWTS program was held in Columbus on 9 to 11 December, 1980, with 763 persons in attendance. There were 130 technical papers and presentations covering the FY 80 activities of the program.

Major consultation activities of the quarter were briefings on the NWTS program to the Governors and their staffs in Maine and Vermont.

Four publications for lay persons which describe the NWTS program were submitted to DOE for review: *NWTS Background Information Pamphlet*, *NWTS Brochure*, *Nuclear Waste Isolation Activities Report*, and *Illustrated Mechanics of Nuclear Waste Isolation*.

A new public information film on the NWTS program was premiered. It is designed to introduce lay audiences to the problem of nuclear waste, describe the mined geologic disposal concept, and show various field research investigations.

WBS 1.5.4

Project: Social Science Research and Support Services

Principal Investigator: Human Affairs Research Centers (HARC) (F. A. Morris)

ONWI Project Manager: J. M. Mountain

Objective

The objective is to build on previous work in social and institutional analysis to help ONWI better address nontechnical concerns in its program planning activities. Specific objectives for FY 81 include (1) further identification of repository siting problems and elaboration of consultation and concurrence, (2) development of plans to respond to regulatory requirements with respect to social and institutional issues, and (3) analysis of the basis for establishing long-term understanding and agreements among government agencies regarding waste management.

Progress Reported Previously

FY 81 is the fifth year in a program of HARC social science support to the NWTS program. In addition to the 12 topical reports prepared through FY 80, HARC has provided substantial assistance to ONWI and DOE in evaluating cross-statements in the NRC Confidence Rulemaking proceedings and in preparing DOE's cross-statement.

Activities During the Reporting Period

The National Site Characterization and Selection Plan was revised by HARC, ONWI, and DOE. Research, consisting of document review, was directed toward preparing internal working drafts and draft topical reports.

SUMMARY

1.6 TEST FACILITIES AND EXCAVATIONS

The scope of the test facilities and excavations task covers the acquisition, design, construction and development, and operation and maintenance of required test facilities and excavations. The prime objective of this task is to provide proper testing facilities to support the test activities of the nuclear waste isolation program.

The Salt Test Facility (STF) Conceptual Design project was organized and initiated. The project scope and roles of the participants in the project have been defined, and progress so far has been reviewed by ONWI and DOE. ONWI received a draft of the STF Functional Design Criteria from Stearns-Roger for review and comment.

A detailed and comprehensive program plan on Alternate Geology Test Facility Mine Screening was submitted on 15 December, 1980. The USBM mineral computer data systems were reviewed by the principal investigator and the project manager to develop a data base of mines that are contained within granitic masses.

WBS 1.6.1

Project: Salt Test Facility Conceptual Design

Principal Investigator: Stearns-Roger Services Inc. (J. C. Mattern)

ONWI Project Manager: J. V. Baum

Objective

The objectives of this project are to:

- Develop a conceptual design for a salt test facility in a dome salt formation or a bedded salt formation
- Prepare cost estimates for the detailed engineering, construction, and decommissioning of the facility
- Prepare an environmental assessment for the facility
- Develop a Quality Assurance Plan for the Title I phase.

The specific locations for the facility will be selected by TBEG. Experimental test requirements will be defined by ONWI and the Westinghouse Advanced Energy Systems Division.

Progress Reported Previously

This project began in November, 1980. Therefore, this is the first quarterly technical progress report.

Activities During the Reporting Period

Major activities were associated with getting the project organized and initiated. The project scope and the roles of the participants in the project have been defined, and the progress of the project has been reviewed.

A draft of the Final Design Criteria was submitted to ONWI for comment.

WBS 1.6.2

Project: Alternate Geology Test Facility—Mine Screen

Principal Investigator: Pincock, Allen & Holt, Inc. (J. R. Swaisgood and L. Kunze)

ONWI Project Manager: M. W. Anthony

Objective

The principal purpose of this project is to identify suitable candidate sites for an Alternate Geology Test Facility (AGTF) in geologic media other than salt. Specific objectives include: (1) compiling a list or data bank of existing underground openings in suitable rock types which meet general identification criteria; and (2) evaluating and comparing candidate sites in granitic rocks using site-specific criteria to identify those sites with sufficient potential to warrant further investigation. The project will be performed using publicly available information only.

Progress Reported Previously

The project was authorized on 12 November 1980, with a contract date of 1 October 1980.

Activities During the Reporting Period

The progress achieved to date consists of a detailed and comprehensive program plan which was submitted on 15 December 1980. The CRIB and MILS computer data systems have been reviewed.

The review was conducted by the principal investigator and the project manager and consisted of discussions with the owners of the data bases and a review of reports which used these data.

Future Activities

Activities for the next reporting period will consist of the start and completion of the following tasks:

- Development and approval of the screening and ranking criteria
- Identification of potential sites for the AGTF
- Ranking of granitic sites will be started in this reporting period.

ABBREVIATIONS

AEC	U.S. Atomic Energy Commission
AEGIS	Assessment of Effectiveness of Geologic Isolation Systems program (program replaces WISAP)
AFR	away from reactor
ANL	Argonne National Laboratory
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASTM	American Society for Testing Materials
BCD	Battelle Columbus Division
BCL	Battelle—Columbus Laboratories
BEG	Bureau of Economic Geology (Texas)
BGI	Bechtel Group, Inc.
BLM	Bureau of Land Management, U.S. Department of the Interior
BNI	Bechtel National, Inc.
BNL	Brookhaven National Laboratory
BNW	Battelle—Northwest
BPMD	Battelle Project Management Division
BPNL	Battelle Pacific Northwest Laboratories
BWIP	Basalt Waste Isolation Project
BWR	boiling water reactor
CANDU	pressurized heavy water reactor (Canadian)
CFR	Code of Federal Regulations
Ci	Curie
CSM	Colorado School of Mines (Golden, CO)
D/D; D&D	decontamination and decommissioning
DOD	U.S. Department of Defense
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
EIS	environmental impact statement
EMAD	Engine Maintenance, Assembly, and Disassembly Facility
EPA	U.S. Environmental Protection Agency
ERDA	U.S. Energy Research and Development Administration
FR	<i>Federal Register</i>
FRG	Federal Republic of Germany
FY (TD)	fiscal year (to date)
GIR	Gulf Interior Region
GPM	Geologic Project Manager
HARC	Human Affairs Research Centers (Battelle—Seattle)
HEDL	Hanford Engineering Development Laboratory
HLW	high-level waste
HWR	heavy water reactor

IAEA	International Atomic Energy Agency
ILW	intermediate-level waste
IWG	Interface Working Group
KBS	Kärnbränslesäkerhet, Swedish Nuclear Fuel Supply Co., Safety Board Division
LASL	Los Alamos Scientific Laboratory
LATA	Los Alamos Technical Associates, Inc.
LBL	Lawrence Berkeley Laboratory
LETCo	Law Engineering Testing Company
LLL	Lawrence Livermore Laboratory
LLW	low-level waste
LSU	Louisiana State University
LVDT	linear variable differential transformer
LWR	light water reactor
MIT	Massachusetts Institute of Technology
NASA	National Aeronautics and Space Administration
NBS	National Bureau of Standards
NEA	Nuclear Energy Agency; National Energy Act
NEPA	National Environmental Policy Act; National Environmental Protection Agency
NOAA	National Oceanographic and Atmospheric Administration
NPO	NWTS Program Office (formerly RL-C)
NRC	U.S. Nuclear Regulatory Commission
NTS	Nevada Test Site
NUS	NUS Corp. (Nuclear Utilities Services Corporation)
NWTS	National Waste Terminal Storage (program)
ONWI	Office of Nuclear Waste Isolation
ORNL	Oak Ridge National Laboratory
OSHA	Occupational Safety and Health Administration
OWI	Office of Waste Isolation (predecessor of ONWI)
PNL	Pacific Northwest Laboratory (Battelle)
PSU	Pennsylvania State University
PWR	pressurized water reactor
QA	quality assurance
R&D	research and development
RHO	Rockwell Hanford Operations
RL-C	Richland Operations, Columbus Program Office, now NPO
SAC	Swedish-American Cooperative
SAI	Science Applications, Inc.
SCEPTER	Systematic Comprehensive Evaluation of Performance and Total Effectiveness of Repositories
SF	spent fuel
SL	Sandia Laboratories
SRL	Savannah River Laboratory
SRP	Savannah River Plant

TASC	The Analytic Sciences Corporation
TBEG	Texas Bureau of Economic Geology
TRU	transuranic (contaminated) waste
USBM	U.S. Bureau of Mines
USGS	U.S. Geological Survey
WBS	Work Breakdown Structure
WES	Waterways Experiment Station, Corps of Engineers
WIPP	Waste Isolation Pilot Plant
WISAP	Waste Isolation Safety Assessment Program (replaced by AEGIS and WRIT)
WRIT	Waste-Rock Interactions Technology program (program replaces WISAP)

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