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SALT REPOSITORY PROJECT CLOSEOUT STATUS REPORT

June 1988

Prepared for

U.S. Department of Energy
Salt Repository Project Office

by

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INTRODUCTION

This report provides an overview of the scope and status of the U.S. Department of Energy's (DOE's) Salt Repository Project (SRP) at the time when the project was terminated by the Nuclear Waste Policy Amendments Act of 1987.* The report can be used as a "roadmap" into the 10-year program of siting a geologic repository for high-level nuclear waste in rock salt formations. Its purpose is to aid persons interested in the information developed during the course of this effort. To assist these persons, each area is briefly described and the major items of information are noted. This report, the three salt Environmental Assessments, and the Site Characterization Plan are the suggested starting points for any search of the literature and information developed by the program participants.

Prior to termination, DOE was preparing to characterize three candidate sites for the first mined geologic repository for the permanent disposal of high-level nuclear waste. The sites were in

- Nevada, a site in volcanic tuff
- Texas, a site in bedded salt (halite)
- Washington, a site in basalt.

These sites, identified by the screening process described in Chapter 3, were selected from the nine potentially acceptable sites shown on Figure I-1. These sites were identified in accordance with provisions of the Nuclear Waste Policy Act of 1982.

DOE's Office of Civilian Radioactive Waste Management, through the Chicago Operations Office, gave authority to the Salt Repository Project Office, together with its integrating contractor, Battelle Project Management Division, to aid DOE in fulfilling its mission. DOE's mission, by direction of the Nuclear Waste Policy Act of 1982, was to provide for the development of repositories for the disposal of high-level radioactive waste and spent fuel in a manner that fully protects the health and safety of the public and the quality of the environment. The mission included four specific goals:

1. Selection and characterization of potential sites
2. Construction and operation of one or more repositories
3. Achievement of public confidence via State, Indian Tribe, and public participation
4. Development of waste packages that meet regulatory performance requirements.

The salt project was fulfilling its role toward achievement of the overall program mission when the Nuclear Waste Policy Act Amendments of 1987 ended its

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*The Nuclear Waste Policy Amendments Act of 1987 terminated the DOE's Texas salt and Washington basalt projects and mandated that site characterization proceed only at the Nevada tuff site.

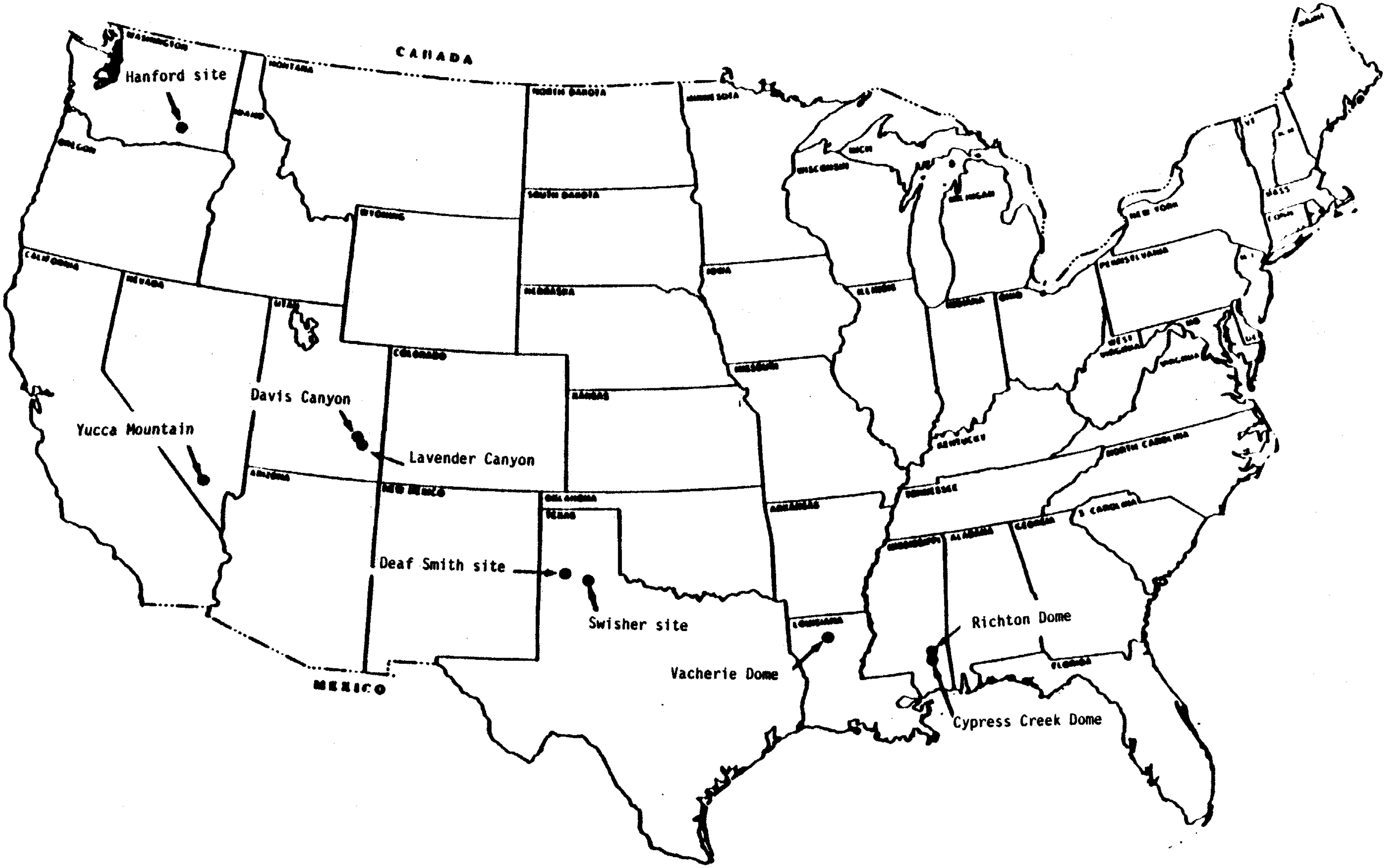


Figure I-1. Potentially Acceptable Sites for the First Repository part of the program. For the goal of selecting and characterizing potential repository sites, the salt project provided nine potentially acceptable sites, three candidate sites (DOE, 1986), and a detailed Site Characterization Plan (DOE, 1988) for the Deaf Smith County site, which is shown in Figure I-2. The goal of constructing and operating one or more repositories would only have been addressed within the SRP if characterization and selection of the Deaf Smith County site had occurred. For the goal of achieving public confidence, the SRP's public information program at potential salt sites operated nine public information offices and held over 1,000 public meetings and briefings for community and organizational leaders in Mississippi, Louisiana, Texas, and Utah during the 10-year program. The SRP selected a candidate waste package and package material early for testing and analysis and was well along in achieving this fourth goal of the program's mission.

To accomplish the above, management of the project responded to program and legislative changes in mission and objectives over the course of 10 years. In May of 1978, Battelle formed the Office of Nuclear Waste Isolation, and DOE formed the National Waste Terminal Storage Program Office (NPO) in Columbus, Ohio, to develop the technology and facilities for the safe disposal of high-level radioactive waste and spent fuel. These offices accelerated the tasks (started in 1976 by DOE and Union Carbide) of screening the country for potentially suitable sites (outside of the Federal Nevada Test Site and Hanford Site) and of bolstering technology of waste forms and techniques of performance assessment. The NPO shed its NWTS program integration responsibilities in 1982 and later became the Salt Repository Project Office (SRPO). The passage of the Nuclear Waste Policy Act Amendments in December 1987 led to close-out of site-specific work in salt by March 1988 and to focusing waste program resources on the Nevada site.

The management responsibility, authority, and accountability for this project was specified in "Charter, National Waste Terminal Storage Program Office" dated December 16, 1981. Responsibilities were amended in "Project Charter for the Salt Repository Project" dated January 16, 1987.

DOE's responsibility was implemented using formal management systems including a work breakdown structure (WBS). This report is structured to parallel the major WBS elements. Chapters 1 through 9 correspond with WBS elements 1.0 through 9.0. Major management and technical contributions to the progress of meeting the program mission and goals are listed below by WBS element. Each chapter describes what was done and the status of the planned work.

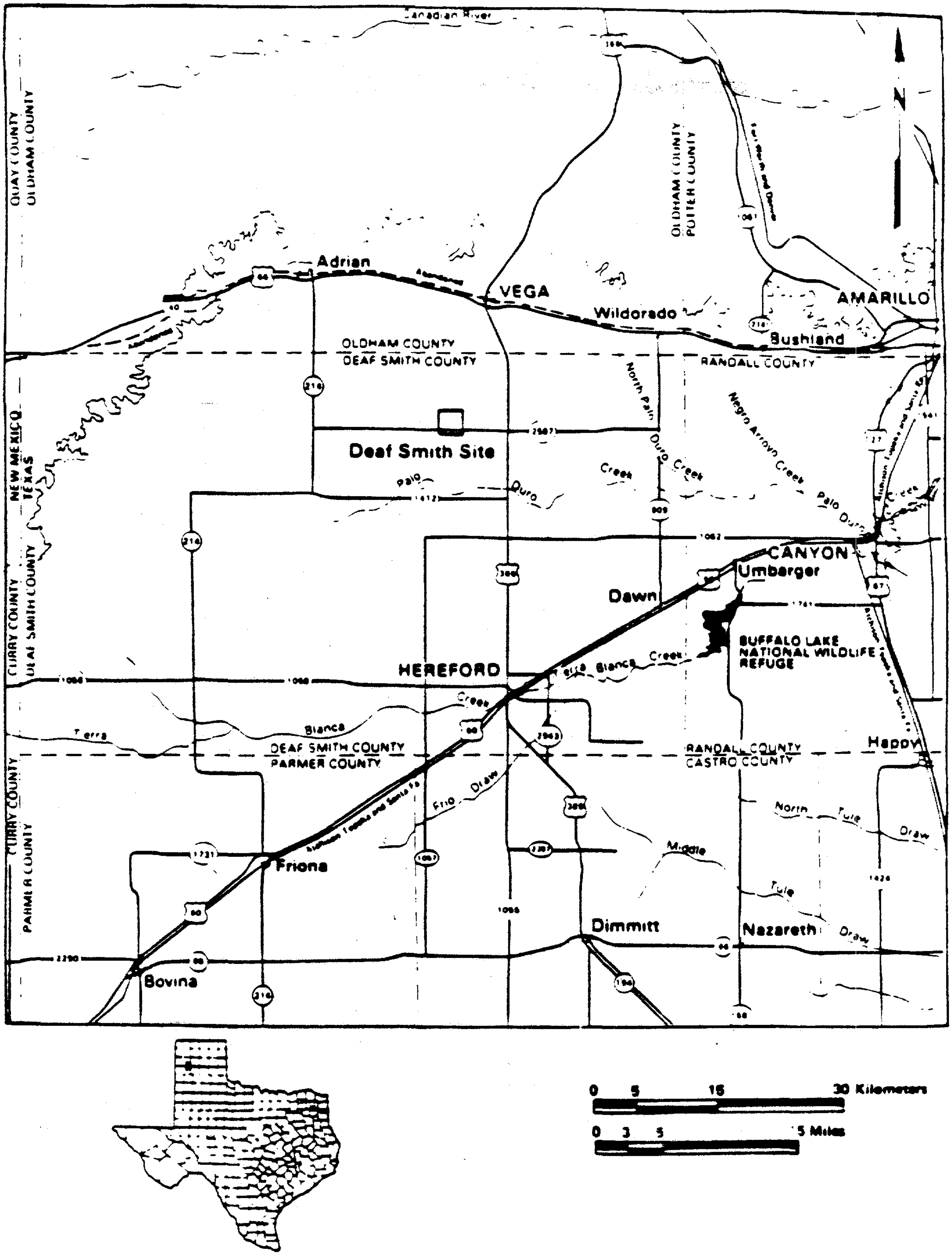


Figure I-2. Deaf Smith County Site, Texas

WBS 1.0 - Systems (Chapter 1)

- Defined the geologic disposal system in bedded salt as shown in Figure I-3
- Implemented a comprehensive systems engineering management plan, requirements document, and technical baselining system
- Developed and documented over three dozen computer codes for the assessment of preclosure and postclosure performance and five computerized repository system cost models.

WBS 2.0 - Waste Package (Chapter 2)

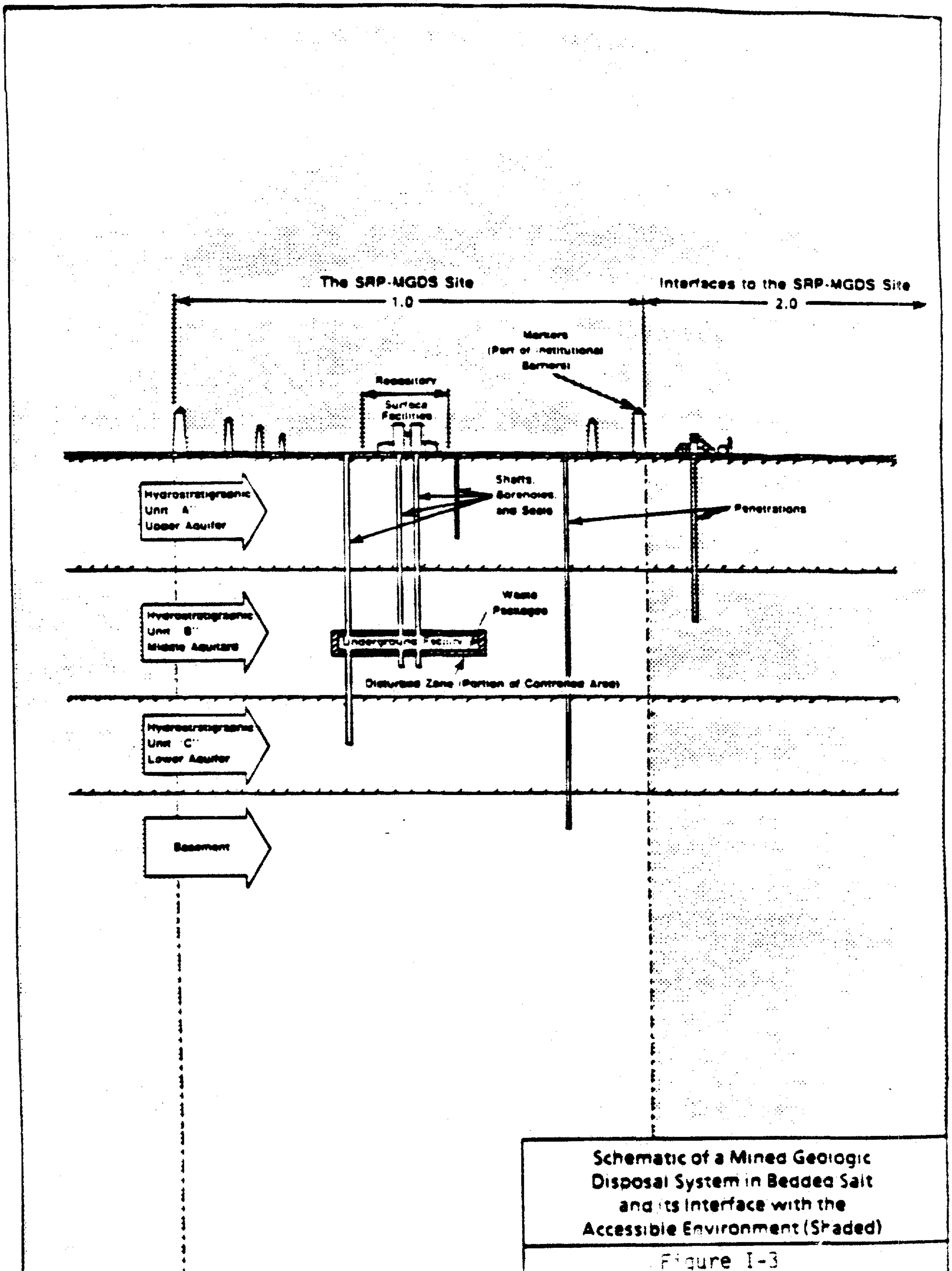
- Developed a conceptual design for the waste package
- Conducted testing and analysis of waste package materials and performance leading to demonstration of a 300- to 1,000-year package lifetime.

WBS 3.0 - Site (Chapter 3)

- Selected three potential sites in salt located in Utah, Mississippi, and Texas and began characterization-phase work at the Deaf Smith County site in Texas (see Figure I-1)
- Drilled and cored scores of boreholes in Utah, Texas, Mississippi, and Louisiana and operated multi-station microseismic networks in southeastern Utah and in the Texas Panhandle
- Developed the comprehensive Socioeconomic Assessment of Repository System (SEARS) model and computer code
- Developed study plans and technical procedures for characterizing the Deaf Smith County site.

WBS 4.0 - Repository (Chapter 4)

- Completed four conceptual repository designs and supporting documents
- Directed preparation of a design guide for repository and exploratory shafts--the first in the world to consider seismic and 100-year design life to exacting quality assurance requirements
- Conducted a major test program in rock mechanics of salt and sealing materials; prepared sealing designs; developed three-dimensional structural models for underground openings.



Schematic of a Mined Geologic Disposal System in Bedded Salt and its Interface with the Accessible Environment (Shaded)

Figure I-3

WBS 5.0 - Regulatory and Institutional (Chapter 5)

- Developed a comprehensive Site Characterization Plan in fulfillment of provisions of the Nuclear Waste Policy Act

- Prepared seven draft and three final Environmental Assessments in support of the DOE Secretary's decision of which sites to characterize and held more than 50 hearings and briefings to encourage public input

- Developed and implemented a four-state public information program and public participation plan

- Developed and implemented Interagency Agreements with the U.S. Geological Survey, the U.S. Army Corps of Engineers, Bureau of Land Management, the National Park Service, and the Bureau of Mines

- Implemented grant programs with the States of Louisiana, Texas, Mississippi, and Utah and with the American Nuclear Society.

WBS 6.0 - Exploratory Shaft (Chapter 6)

- Completed the final design for an exploratory shaft facility

- Developed an innovative process for design reviews.

WBS 7.0 - Test Facilities (Chapter 7)

- Conducted joint experiments with the German program at the Asse salt mine to study combined radiation, thermal, and mechanical effects

- Developed a comprehensive plan for conducting underground tests in the Exploratory Shaft Facility.

WBS 8.0 - Land Acquisition (Chapter 8)

- Completed land acquisition planning efforts that were fully coordinated with the U.S. Army Corps of Engineers.

WBS 9.0 - Project Management (Chapter 9)

- Handled procurement for and managed the work of over 400 firms, consultants, and national laboratories, and developed a DOE-certified procurement system and project management and control system for this work

- Developed a Quality Assurance Program that was instrumental in forming overall QA policy and procedures for the whole program.

These tasks received great public and agency scrutiny and were subject to independent peer reviews. The National Academy of Sciences, for example, performed an independent appraisal of the certainty with which repository system performance could be predicted. Thousands of comments from Federal agencies, State agencies, and the public were addressed. A Program Review Committee provided a periodic assessment of how the program was perceived by the public on critical technical and programmatic issues. Community leaders and the public helped develop a public participation plan.

References follow each chapter. For an indexed bibliography of SRP studies between April 1978 and May 1986, see:

DOE (U.S. Department of Energy), 1986. Bibliography of Studies for the Salt Repository Project Office of the Civilian Radioactive Waste Management Program April 1978 - May 1986, DOE/CH/10140-05(86), prepared by the Office of Nuclear Waste Management, Battelle Project Management Division, Columbus, Ohio, for U.S. Department of Energy, Washington, D.C.

In addition, many of the documents referenced in this report contain extensive bibliographies of work performed. Copies of referenced reports published as part of the Salt Repository Project may be requested from the National Technical Information Service (U.S. Department of Commerce, 5285 Port Royal Road, Springfield, VA 22161). Copies of referenced draft reports that had not yet been published at the time of project termination in December 1987 may be requested from the Office of Scientific and Technical Information (U.S. Department of Energy, P.O. Box 62, Oak Ridge, TN 37831). Access to project records may be obtained by contacting the Office of Civilian Radioactive Waste Management, U.S. Department of Energy (Room 5-A-085 Forrestal Building, 1000 Independence Avenue, S.W., Washington, DC 20585).

The following chapters describe what has occurred in the Salt Repository Project.

References

DOE (U.S. Department of Energy), 1985. Mission Plan for the Civilian Radioactive Waste Management Program, DOE/RW-0005, Office of Civilian Radioactive Waste Management, Washington, DC.

DOE (U.S. Department of Energy), 1988. Site Characterization Plan, Deaf Smith County Site, Texas, Consultation Draft Nuclear Waste Policy Act (Section 113), DOE/RW-0162, U.S. Department of Energy, Washington, DC.

CHAPTER 1

SYSTEMS (WBS 1.0)

The objectives of WBS 1.0 - Systems included the following:

- Integration of project-wide systems and task activities toward meeting the project mission and assessment of the progress toward achievement of that goal
- Development and application of systems engineering methods to facilitate the development, operation, and decommissioning of a mined geologic disposal system
- Establishment, operation, and maintenance of a project-wide base of significant technical data
- Assessment of the performance of the mined geologic disposal system.

The work performed by the project to accomplish the above objectives was addressed under the following work breakdown structure subelements:

- WBS 1.1 - Management and Integration
- WBS 1.2 - Systems Engineering
- WBS 1.3 - Technical Data Base Management
- WBS 1.4 - Total System Performance Assessment.

WBS 1.1 - Management and Integration concerned the management, integration, and assessment of various project activities. A discussion of the project work in this area is provided in Section 1.1.

WBS 1.2 - Systems Engineering concerned the development and application of the systems engineering process to effect the orderly evolution of the mined geologic disposal system. This work encompassed systems studies and analyses; system requirements, descriptions, and costs; configuration management; and the SRP transportation program. A discussion of the project work in systems engineering is provided in Section 1.2.

WBS 1.3 - Technical Data Base Management concerned the establishment and maintenance of a base of significant technical data that would be used in evaluating the performance of the waste isolation system. A discussion of the project work in this area is provided in Section 1.3.

WBS 1.4 - Total System Performance Assessment included both preclosure and postclosure performance assessments and encompassed the total waste isolation system. This work supported the environmental assessment reports, the site characterization report, and the safety analysis reports and included recommendations relative to site characterization and design data needs. This work also included the development, documentation, verification, and validation of performance assessment computer codes. A discussion of performance assessment is provided in Section 1.4.

1.1 MANAGEMENT AND INTEGRATION (WBS 1.1)

The management and integration of Systems concerned the various project activities conducted under WBS 1.0 and included the coordination of Systems' activities with DOE and the other OCRWM projects. This work typically involved such activities as budgeting and scheduling of tasks, meeting planning and coordination, technical presentations, technical and programmatic document review, etc. A discussion of these rather routine management and integration activities is not provided. However, the following two items were significant in the effective accomplishment of the management and integration function:

- Systems Engineering Management Plan

- Project Networks.

1.1.1 Systems Engineering Management Plan

Systems engineering is a process to effect the orderly evolution of a system from the point when a need is identified, through the design, construction and operation stage, to the ultimate phase-out of the system. DOE Order 4700.1, Project Management System, recommends that the systems engineering process be adopted by any project requiring an integrated engineering procedure. OCRWM directed that systems engineering techniques be implemented throughout the waste management program. Accordingly, to prescribe the systems engineering procedures to be utilized at each level of the program, Systems Engineering Management Plans (SEMPs) were developed by OCRWM, OGR, and SRP.

The SRP SEMP was published in August 1986 as DOE/CH-21 (SRPO, 1986b). It has two principal purposes. First, it prescribes how systems engineering will be implemented by the SRP at the project level. Second, it identifies the minimum requirements for a systems engineering approach at the contractor level.

The document includes an introduction and an explanation of the SEMP in terms of DOE directives. It describes the strategy for conducting systems engineering on the SRP and identifies the positions within the project organization that are responsible for carrying out these systems engineering activities. It also describes project management tools, including plans, reviews, and documentation, that are used to implement the systems engineering process.

The SRP SEMP is a controlled document in the SRP baseline identified as SRP/B-12. As of January 1988, no changes to the SEMP had been approved by the SRP Change Control Board, although several changes were being considered.

1.1.2 Project Networks

Activity networks were used for the planning and control of the project. The networks were used to show the sequencing of activities that must be accomplished and the relationships between various activities in the project.

Under WBS 1.0 - Systems, project activity networks were developed to a level corresponding to the third level of the work breakdown structure; in selected cases fourth- and fifth-level networks were developed. The major emphasis during this network development was on the network logic: identifying the activities to be performed and the relationships between activities.

After development, the responsibility for the project activity networks was transferred to WBS 9.0 - Program Control. For a discussion of the use of the networks in project budgeting and scheduling, see Chapter 9 of this report.

1.2 SYSTEMS ENGINEERING (WBS 1.2)

Systems engineering was defined earlier as a process to effect the orderly evolution of a system. Some of the activities that contributed to the systems engineering process were performed under work breakdown structure subelement WBS 1.2. These activities included the following:

- Systems Studies
- System Analysis Tools
- Total System Life-Cycle Costs
- SRP Requirements and System Description
- Configuration Management System
- SRP Transportation Program

and are addressed below.

1.2.1 System Studies

Although requirements for an SRP system studies register and systems study reports are given in the SRP SEMP, a formal systems studies register was not published and few systems study reports were ever published. Nevertheless, various studies were performed by the project to better define the mined geologic disposal system and to identify project and system requirements. In 1986, a draft systems studies register was compiled based on discussions with the project technical staff. This draft listed a host of studies being considered or being performed in the following four categories:

- Interface Requirements/Requirements Allocation
- Definition of Terms

- Adequacy
- Trade-Off Studies/Decision Analysis.

These studies are identified below, with the status updated to January 1988. Note that most studies related to the repository are in an on-going status because they were to have continued through the advanced conceptual design phase.

Interface Requirements/Requirements Allocation

Title: Transportation/Receiving Facility

Organization: ONWI

Scope: The scope covers repository receiving facilities and operations from the point of shipment acceptance from the carrier at the entrance (security gate) to the protected area, to the hot cell unloading port, and return of the empty transportation system components to the carrier at the exit from the protected area.

Status: Study ongoing. For progress see the following unpublished reports by ONWI: Peterson, R. W., et al., "Operational Analysis Supporting the Definition of the Repository/Transportation Interface," October 1985; and Smith, L. A., et al., "Salt Repository Project Transportation System Requirements: Transportation System/Repository Receiving Facility Interface Requirements," January 1988. See also Chapter 4 of this report.

Title: Waste Package/Repository

Organization: ONWI

Scope: Define the design requirements for the repository to interface with the waste package. Interfaces include production rates, sizes, weights, radiation levels, materials of construction, waste payload, etc.

Status: Study ongoing. For progress see "Waste Package Repository Impact" (Fluor, 1986c). See also Chapters 2 and 4 of this report.

Title: Retrievability

Organization: ONWI/FLUOR

Scope: Complete analyses and testing to support the SRP position on retrieval.

Status: Study ongoing. For progress see the following report by Tomé et al.: Salt Repository Project Retrieval Strategy Study Draft, December 1987. See also Chapter 4 of this report.

Title: ESF/Repository

Organization: ONWI/FLUOR/PB

Scope: Identify the impacts of incorporating the ESF into the repository. Address several ESF cases and assess their impacts on repository design, construction schedule, and decommissioning. Also, define the design requirements for the repository to interface with the ESF. Interface includes site constraints, ability to isolate ESF from repository, and impact of ESF operation concurrent with repository development and emplacement operations.

Status: Study ongoing. A major product of this work was the SRP Shaft Design Guide, 1987, developed by Fluor and PB/PB-KBB. See also Chapter 4 of this report.

Title: MRS/Repository

Organization: FLUOR

Scope: Assess several cases involving range of MRS and repository operating constraints. Results will be used to optimize overall waste management system, including design of MRS and repository. Results will also support DOE preparation of MRS proposal to Congress.

Status: Study completed. For results see the unpublished report "Monitored Retrievable Storage Facility and Salt Repository Integration" (Fluor, 1986b).

Title: Backfilling/Emplacement Operations

Organization: FLUOR

Scope: Perform trade-off study between repository emplacement and backfill operations. Results to be used to optimize repository design. Impacts include waste emplacement schedule, room stability, retrievability, and decommissioning.

Status: Study not initiated.

Title: Accident Scenarios

Organization: ONWI/FLUOR

Scope: To assess accident conditions on the basis of scenarios selected for each facility component, its characteristics and compliance with requirements. To identify systems, structures, and components which are important to safety.

Status: Study ongoing. For progress see the following unpublished documents by ONWI: "Probabilistic Risk Assessment for Salt Repository Conceptual Design of Subsurface Facilities: A Technical Basis for Q-List Determination" (1987d) and "Salt Repository Project Q-List Working Group Report" (1987e). See also the Fluor SCP-CDR, 1987.

Title: Engineered Barrier System

Organization: ONWI/SAIC

Scope: Utilize verified and validated computer models for waste package, engineered barrier system, and site to show compliance with NRC requirements, and to perform total system analysis to show compliance with EPA requirements.

Status: Study ongoing. For early progress, see Systems Study on Engineered Barriers: Barrier Performance Analysis, ONWI-211, September 1980, by R. T. Stula et al. See also Chapter 2 of this report.

Title: Annual Exposure Dose/Radiation/Nonradiological Safety Goal

Organization: ONWI

Scope: Specify or quantify general approaches specified in performance assessment plan.

Status: Study completed. For results see Preclosure Radiological Calculations to Support Salt Site Evaluations, BMI/ONWI-541 (D. A Waite, 1986) and BMI/ONWI-541 (Revision 1), (D. A. Waite et al., 1987).

Title: Underground Facility/Emplacement Operations

Organization: FLUOR

Scope: Assess operating concepts for repository development and emplacement operations. Define interfaces in areas such as ventilation, salt handling, waste emplacement, men and equipment utilization, etc.

Status: Study not initiated.

Title: Waste Processing/Mine Emplacement Study

Organization: FLUOR

Scope: Define the repository subsystems in terms of functions and flow diagrams. Results include defining subsystem interfaces, system throughput, storage requirements, areas of development, etc.

Status: Study ongoing. See the Fluor SCP-CDR, 1987.

Title: Environmental/Socioeconomic Constraints

Organization: ONWI

Scope: Establish environmental design criteria and environmental/socioeconomic constraints.

Status: Study ongoing. For progress see the unpublished ONWI documents "Socioeconomic Monitoring and Mitigation Plan" (1987f) and "Environmental Regulatory Compliance Plan" (1987a).

Definition of Terms

Title: Reasonably Available Technology

Organization: ONWI

Scope: Identify options for definition and select the preferred option with rationale for selection.

Status: Study not initiated.

Title: Utilization of and Acceptability Criteria for "Old" Data

Organization: ONWI

Scope: Provides the information on the process of accepting data prior to its utilization in the licensing documents.

Status: Study not initiated.

Title: ALARA

Organization: ONWI/FLUOR

Scope: Evaluate alternative definitions and establish criteria by which the designs can be evaluated to show conformance with the requirements.

Status: Study not initiated.

Title: Underground Facility/EBS Boundary, Disturbed Zone

Organization: ONWI

Scope: Identify several options for definitions, and select the preferred option with rationale for selection.

Status: Study completed. For results see DOE Memorandum by R. Stein, "DOE Interpretation of Terms in 10CFR60" (DOE, 1987).

Title: Substantially Complete Containment

Organization: ONWI

Scope: Evaluate alternative definitions and develop rationale for selection of a specific definition. Establish criteria by which to determine compliance.

Status: Study completed. For results see DOE Memorandum by R. Stein, "DOE Interpretation of Terms in 10CFR60" (DOE, 1987).

Title: Interpretation of Regulatory Requirements

Organization: ONWI/EBASCO

Scope: Provide regulatory intent and salt specific interpretations to analyze regulatory requirements as they apply to characterization and design and identify the method to demonstrate conformance.

Status: Study completed.

Title: Reasonable Assurance

Organization: ONWI

Scope: To research NRC's approach to "reasonable assurance" that has been used previously in the licensing process and establish criteria by which to evaluate the design.

Status: Study not initiated.

Adequacy

Title: Scenario Selection Criteria

Organization: ONWI

Scope: Develop scenarios and perform analyses to show compliance with regulations for natural processes such as climate and tectonic, and human-induced effects such as drilling.

Status: Study ongoing. For progress see Preliminary Analyses of Scenarios for Potential Human Interference for Repositories in Three Salt Formations, BMI/ONWI-553, October 1985, by INTERA. See also Section 1.4 of this report.

Title: Sensitivity/Uncertainty Methods and Applications

Organization: ONWI/ORNL/INTERA/PNL/BCD

Scope: To identify the key parameters of the overall systems model (brine migration rates, solubility of radionuclides, permeability) by integrating existing subsystem codes and analyzing cause/effect relationships from a statistical standpoint.

Status: Study ongoing. See Section 1.4.4 of this report.

Title: Operational Definition of Demo Testing Requirements

Organization: ONWI

Scope: To interpret the regulation 10 CFR Part 60 and define testing requirements to structure the in situ test program.

Status: Study completed. For results see Draft Underground Test Plan for Site Characterization and Testing in an Exploratory Shaft Facility in Salt, BMI/ONWI-644, May 1987, by Golder. See also the DOE SRP SCP, 1988.

Trade-Off Studies/Decision Analysis

Title: Salt Management Options

Organization: FLUOR

Scope: Perform trade-off study to optimize excavated salt handling operations, assess impacts on subsystems such as surface handling facilities, room stability, shaft use, development requirements in advance of emplacement, etc.

Status: Study ongoing. For progress see the Fluor SCP-CDR, 1987, and Chapter 4 of this report.

Title: Ventilation-Retrievability Options

Organization: FLUOR

Scope: Define ventilation requirements for postulated retrieval scenarios. Determine capability of existing design concepts versus need for additional capacity in terms of design and procurement constraints, etc.

Status: Study ongoing. For progress see the report: Salt Repository Project Retrieval Strategy Study Draft (Tomé et al., 1987).

Title: Shaft Sizes

Organization: FLUOR

Scope: Define requirements for determining sizes and quantities of shafts in the repository. Trade-off studies are required to assess construction, operation, and decommissioning impacts of shaft sizes.

Status: Study not initiated.

Title: Subsurface Operation Rates

Organization: FLUOR

Scope: Define operating concepts by considering interfaces and requirements for development, emplacement, salt handling, and waste disposal operations. Trade-off studies are required to assess impacts of alternative excavation/emplacement processes. Results include optimized concepts, e.g., staffing, equipment, costs, and schedule.

Status: Study not initiated.

Title: Horizontal versus Vertical Emplacement

Organization: FLUOR

Scope: Evaluate various alternative waste emplacement modes in terms of design requirements and minimization of adverse consequences. Define advantages and disadvantages of various modes and determine optimum choice.

Status: Study ongoing. For results see "Salt Repoitory Project Waste Emplacement Mode Decision Paper" (SRPO, 1987h). See also Chapter 4 of this report.

Title: Cross-Cut Spacing versus Methane Concentration

Organization: FLUOR

Scope: Evaluate existing gassy mine regulations in terms of cross-cut spacing requirements. Perform trade-off study of various subsurface layouts to determine impact of potential gassy condition on design.

Status: Study ongoing. For progress see the unpublished report "Applicability of Gassy Mine Regulations" (Fluor, 1986a).

Title: GROA Surface/Subsurface Area Systems Operations Performance Analyses

Organization: ONWI/FLUOR

Scope: Utilize computer simulation models of repository operations to support design trade-off studies.

Status: Study ongoing. For progress see the unpublished report by Fluor: "Repository Simulation Model Final Report," 1988. See also the Fluor SCP-CDR, 1987.

Title: Gassy Mine Requirements versus In Situ Testing

Organization: ONWI

Scope: The study provides the interpretations of the regulations and their effect on ESF design and in situ testing, and the variances which may be required.

Status: Study ongoing.

Title: System Performance Analyses

Organization: ONWI

Scope: Analyses utilizing system models which will evaluate systems designs against the NRC and EPA regulations.

Status: Study ongoing. See Section 1.4 of this report.

Title: Type of Construction of Surface Facilities

Organization: ONWI

Scope: Analyze and compare life-cycle costs of surface facilities constructed of different materials, to support ESF design.

Status: Study ongoing. For progress see the unpublished ESF Title I and Title II Design Reports by PB/PB-KBB (1987, 1988).

Title: Sensitivity/Uncertainty Analyses of Performance

Organization: ONWI/ORNL

Scope: Develop methods and carry out the analyses to show compliance with regulations considering uncertainties.

Status: Study ongoing. See Section 1.4.4 of this report.

Title: Design Implications of Reducing or Increasing Exposure

Organization: ONWI/FLUOR

Scope: Research the cost implication of dose limit changes.

Status: Study not initiated.

1.2.2 Systems Analysis Tools

Various tools for system analysis or systems engineering have been used on the project. Two tools developed for use on the project merit special attention and are discussed below. These tools are the SCP Parameters List, and simulation modeling.

1.2.2.1 SCP Parameters List

The SCP Parameters List is a methodology developed by the project to facilitate and to improve the integration of the Site Characterization Plan. This methodology consisted of applying a process that uses a set of specific, unique, and quantifiable parameters as the foundation to define relationships among the parameters and to define relationships among related entities such as study plans, requirements, and issues. A computerized knowledge-based program was used to sort the information and to assist in the analysis of the information.

The key to the process was to develop a set of parameters that, taken in toto, characterizes the waste disposal system. A parameter, as described in the performance allocation process, is a multi-tiered characteristic of a specific system element. The term characteristic includes an information need, a general characteristic, and, if necessary, a subcharacteristic. The term system element refers to the physical subsystems and components of a mined geologic disposal system, and includes engineered systems as well as measures of the stratigraphic and areal extent of natural systems. As an example, one parameter in the SCP is given by the general characteristic "Density," and the information need "Fluid Hydraulic Properties," and the system element "Site HSU C." Therefore, the specific parameter identified is density of the fluid contained in "Site HSU C." The parameter supports the information need "Fluid Hydraulic Properties."

The next step was to specify the information base that supports a parameter and the information requirements that justify the parameter. For the SCP, this step involved identifying (1) the SCP section which provides the current knowledge regarding the parameter; (2) the SCP section which addresses the need for more information regarding the parameter; (3) the study, test, or analysis plans that discuss how the needed information will be obtained; (4) the numbered issue, from DOE's Issue Hierarchy, that will be addressed using the parameter; and (5) the general purpose that will be served by the parameter. Continuing with the SCP example of the parameter "Site HSU C/Fluid Hydraulic Properties/Density," this step shows that (1) current knowledge of the parameter is given in SCP Section "4.2"; (2) the need for more information about the parameter is given in SCP Section "8.3.1.7"; (3) the needed information will be obtained from a plan identified as "SRMSSP"; (4) the parameter is need to address Issue "1.1.18"; and (5) the parameter is used "To calculate flow paths, velocities, travel times and to analyze radionuclide transport." It should be noted, however, that a given parameter could have multiple entries in the categories listed above.

During the development of the SCP, individual parameters and their associated information bases and requirements were identified. This identification was provided by a host of researchers from different technical disciplines working on different parts of the SCP. The resulting data base, termed the SCP Parameters List, contained a massive amount of information.

A computerized knowledge-based program was developed to sort the information in the Parameters List and to assist in the analysis of the information. The computer program was written in the language PROLOG and developed on an IBM PC-AT. Using this program it was possible to greatly improve the quality of the Parameters List and enhance the integration of the SCP. For example, ambiguity was reduced by the use of specific terms and definitions; redundancy of parameters was identified and eliminated; parameters lacking a test plan were identified; and test plans which had no identified users or overlapped other test plans were found. In addition, the program allowed the Parameters List to be viewed from many different perspectives; this facilitated examination of the Parameters List by issue and by discipline, and also aided in the examination of the coverage to be provided by study, test, and analysis plans.

No published reports on the SCP Parameters List methodology are available. The usage of the Parameters List can be observed in the DOE SRP SCP. More information on the SCP Parameters List methodology is given in the unpublished report by Korns, D. E., and Troy, K. S., entitled "SCP Tool User's Guide," March 1988.

1.2.2.2 Simulation Modeling

Simulation modeling was used by the project to facilitate the design and analysis of the nuclear waste repository. Simulation modeling provided a method to integrate basic repository operations and to analyze the resulting system performance.

As discussed below, the project was developing three simulation models:

- SIMREP
- SIMMINE
- SIMHAND.

The simulation language used for the model development was SIMSCRIPT, a highly flexible language for discrete-event simulation modeling. Model development was done on an IBM PC-AT; the models were also run on an IBM mainframe for cases requiring extensive calculations.

SIMREP is a discrete-event computer simulation of repository operations in a salt repository waste handling facility. The main purpose of SIMREP is to estimate the development of potentially hazardous queueing bottlenecks within the repository. The basic guidelines and operating logic for SIMREP were provided by Fluor, the architect-engineer for the repository. This logic included all major operations in the waste handling facility resulting from cask arrivals, waste off-loading, disassembly, canistering, overpacking, waste-shaft operations, and underground waste emplacement.

SIMREP models the major functions and operations in the surface waste handling facility. Three types of waste forms are included in the model: (1) spent fuel assemblies, BWR and PWR; (2) reactor-consolidated spent fuel, BWR and PWR; and (3) high-level waste, DHLW and WVHLW. The major processes modeled include (1) truck and train processes; (2) shipping unit process; (3) cask process; (4) container process; (5) SFA and RCSF processes; (6) canister and HLW canister processes; and (7) overpack, transfer cask, hoist, and underground transporter processes.

SIMREP can be used to aid in repository design by addressing questions of resource allocation; for example, determining the number of truck cask unloading stations needed to handle the expected spent fuel throughput without excessive queueing, or determining the operational impacts of surge storage capacities. SIMREP can also be used to help analyze potential operational problems in the area of radiation protection; for example, information on the number of spent fuel casks accumulating in a queue can be used to estimate radiation field intensities.

SIMREP is documented in two published reports: BMI/ONWI-647, Verification Report for SIMREP 1.1 (Tarapore, 1987), and BMI/ONWI-648, SIMREP 1.1 -A Simulation Model for Repository Operations (Tarapore et al., 1987). Addition information on SIMREP can be obtained in the SRP Closeout Task Report by ONWI entitled "Repository Simulation Modeling" (ONWI, 1988a).

SIMMINE is a discrete-event computer simulation of the salt mining operations of a typical subpanel of a nuclear waste repository in salt. The main purpose of SIMMINE is to determine the optimal allocation of the required mining resources for any desired subsurface panel layout. The design logic for the mining operations was provided by Morrison-Knudsen Engineering, the subcontractor to Fluor for mining the salt repository.

SIMMINE models the major functions and operations required for excavating the subsurface waste emplacement rooms (entries). These operations include mining of the salt face and transporting the mined salt to a conveyor belt feeder-breaker system. The major equipment considered includes the miner, the shuttle cars (haulers), the feeder-breaker unit (tailpiece), and the roofbolter. Equipment breakdown and repair features are included in the model.

SIMMINE can be used to aid in the mining system design; for example, determining the optimal forward progress distance of the miner prior to moving the feeder-breaker tailpiece or determining the optimal number of hauler trucks to be allocated to a subpanel. SIMMINE can also be used to address equipment reliability and maintenance issues; for example, the effect of a specified mean-time-between-breakdowns of a miner on system performance could be determined.

A unique feature of SIMMINE is that the model can also represent the mining simulation graphically. This graphics capability is provided by SIMANIMATION, a graphics package available with SIMSCRIPT. This graphics capability proved to be a valuable tool during program development and debugging.

SIMMINE is documented in an unpublished report by J. M. Furr et al. (1987) entitled "SIMMINE 1.0 - A Simulation Model for Subsurface Mining Operations." Additional information on SIMMINE can be obtained in the SRP Closeout Task Report by ONWI entitled "Repository Simulation Modeling."

SIMHAND is a discrete-event computer simulation of a material handling system. The system is for the transportation of mined salt from the mining face, through the conveyor belt system, to the surge storage bin, and then out to the hoist, salt block production facility, or backfill storage area. The design logic for the material handling system was provided by Morrison-Knudsen Engineering, the subcontractor to Fluor for mining the salt repository.

Detailed information on SIMHAND is not currently available; the code is not documented in any presently available reports. The most current information on SIMHAND can be obtained in the SRP Closeout Task Report by ONWI entitled "Repository Simulation Modeling" (ONWI, 1988a).

1.2.3 Total System Life-Cycle Costs

The term system life-cycle cost refers to all costs associated with the system over its entire life cycle. For a mined geologic disposal system, this life cycle includes engineering development, design, construction, system operation, decommissioning, closure, and postclosure monitoring. Life-cycle costing is required for the annual evaluation of the Nuclear Waste Fund fee adequacy.

During the early phase of the project, most cost analysis work was intended to provide waste program participants with a basic knowledge of the economics of mined geologic repositories. Major documents published during this period include: Forster, J. D., The Economics of Mined Geologic Repositories, ONWI-93, December 1979; Brown, R. W., Standardized Repository and Encapsulation Facility Cost Estimates for Comparative Evaluation and Pricing Study, ONWI-110, July 1980; Waddell, J. D., et al., Projected Costs for Mined Geologic Repositories for Disposal of Commercial Nuclear Wastes, ONI-3, December 1982; and Hoffman, P. L., and Dippold, D. G., "The Economics of Mined Geologic Repositories," in CONF-831217, 1983.

Subsequent cost analysis work was more focused on specific aspects of the waste disposal system. The effects of spent fuel burnup and age on repository costs were addressed in: Waddell, J. D., et al., "The Impacts of Waste Age," DOE/NWTS-15, 1981, and Dippold, D. G., and Wampler, J. A., Spent Fuel Burnup and Age: Implications for the Design and Cost of a Waste Disposal System, BMI/ONWI-561, December 1984.

A variety of computerized cost models were developed by the project to perform cost analyses in support of engineering design and systems tradeoff studies. A brief survey of these models is provided below.

- WADCOM. The WADCOM model is a modular computer model designed to allow one to explore and compare the cost-effectiveness of different nuclear waste disposal options. The model is a relatively aggregate representation of the life cycle costs associated with moving, processing, and ultimately disposing of nuclear waste. The model is documented in a published report: Dippold, D. G., et al., WADCOM: A Waste Disposal Cost Model, ONI-6, June 1983.

- PACCOM. The PACCOM model is a computerized, parametric model used to estimate the capital, operating, and decommissioning costs of a variety of nuclear waste packaging facility configurations. The model is based on a facility concept from which functional components of the facility have been identified and their design and costs related to various parameters such as waste type, waste throughput, and the number of operational shifts employed. The functional components which the model considers include hot cells and their supporting facilities, transportation, cask handling facilities, transuranic waste handling facilities, and various general support and administrative facilities. The model is documented in an unpublished report by Dippold, D. G., et al. entitled "PACCOM: A Nuclear Waste Packaging Facility Cost Model," dated May 1985.

Following the passage of the Nuclear Waste Policy Act of 1982 and the introduction of Fluor as the repository architect-engineer, the direction of the life-cycle cost work within the project changed. The role of WBS 1.0 - Systems was to assure the completeness and integration of cost estimates provided by other WBS elements and contractors, e.g., WBS 4.0 - Repository and Fluor, and to provide the SRP Life-Cycle Cost Estimate required for the annual

Nuclear Waste Fund fee adequacy evaluation. This work involved assimilating all project costs into one total project estimate, and reviewing the estimated costs to establish a level of validity of the estimate. In addition, the project maintained a data base of the cost estimates and had the capability to examine the effects of various cost adjustment factors, e.g., engineering cost escalation factor, emplacement wage rate factor, on the total estimated cost.

One aspect of the project's life-cycle cost estimation work that has not yet been discussed is the consideration of uncertainties in cost estimates. The initial work in cost estimate uncertainty analysis is addressed in a paper by Tzemos, S., and Dippold, D., entitled "Stochastic Cost Estimating in Repository Life Cycle Cost Analysis," presented at the Institute of Nuclear Materials Management meeting in New Orleans in June 1986, and in an unpublished report by Thomas, R. E., et al., entitled "Sensitivity and Uncertainty Analysis of Linear Cost Codes," September 1986. Potential difficulties in applying these analysis procedures and in explaining their validity and usefulness led the project in the direction of simpler analysis tools. The computer program RACE, for Risk Analyzer of Cost Estimates, was developed to generate a cumulative distribution function of the expected life-cycle cost using Monte Carlo sampling. RACE was written in the computer language PASCAL and implemented on an IBM PC-AT. RACE was designed to interface with project cost data supplied as an ASCII file and accept user input on component cost probability distributions. The documentation of RACE is sparse; for an update of the status of the code, see the SRP Closeout Task Report by ONWI entitled "Performance Assessment Code Documentation."

1.2.4 SRP Requirements and System Description

One of the major tasks of WBS 1.0 - Systems was the development of program requirements and the allocation of the requirements to systems and subsystems. The accomplishment of the task involved the development of a requirements document and a companion system description document. It is important to note that the requirements document addresses both program requirements and requirements imposed on a mined geologic disposal system, whereas the description document covers only the system, not the entire program.

The initial set of requirements developed by the project was for the National Waste Terminal Storage (NWTS) program and, although focused on a mined geologic repository, was not specifically tailored to a salt site. The requirements were published as "NWTS Program Criteria for Mined Geologic Disposal of Nuclear Waste" in a four-volume set in the DOE/NWTS-33 series, specifically:

- NWTS-33(1), "Program Objectives, Functional Requirements, and System Performance Criteria" (DOE, 1982c)
- NWTS-33(2), "Site Performance Criteria" (DOE, 1981)
- NWTS-33(3), "Repository Performance and Development Criteria" (DOE, 1982d)

- NWTS-33(4a), "Functional Requirements and Performance Criteria for Waste Packages for Solidified High-Level Waste and Spent Fuel" (DOE, 1982b).

Following the passage of the Nuclear Waste Policy Act of 1982, program-level requirements became the responsibility of the Office of Civilian Radioactive Waste Management (OCRWM). In addition, the Office of Geologic Repositories (OGR) was created within OCRWM to direct the media-specific investigations to be carried out by the three projects. Accordingly, the project's work became focussed on the requirements for a mined geologic disposal system in salt.

The development of requirements for a mined geologic disposal system in salt resulted in an unpublished document by ONWI entitled "Salt Generic System Requirements Specification" (SG-SRS), 1985. This document was intended to be the salt-specific interpretation of DOE's Generic Requirements for a Mined Geologic Disposal System (1984). It was not site-specific, but did consider three regions having potentially suitable salt formations: dome salt formations in the Gulf Interior Region and bedded salt formations in the Permian and Paradox Basin Regions. Continuing work on the system requirements specification transformed it into a more comprehensive requirements document.

The continued development of requirements resulted in SRPO's Salt Repository Project Requirements Document (SRP-RD), Revision 4 (1987f). The SRP-RD translated the generic program requirements into detailed project requirements for the development and evaluation and subsequent deployment through construction, operation, closure/decommissioning, and postclosure of a Mined Geologic Disposal System (MGDS) for the disposal of high-level radioactive wastes in a bedded salt formation in Deaf Smith County, Texas. Some key features of the SRP-RD are the following:

1. It is a comprehensive project requirements document, not just a systems requirements document.

2. It covers the life-cycle of the project.

3. It is site-specific.

4. It is consistent with DOE's Generic Requirements for a Mined Geologic Disposal System and addresses Federal (e.g., DOE, EPA, NRC) and State of Texas requirements and regulations.

The SRP-RD is a controlled document in the SRP baseline identified as SRP/B-18. The document was baselined in October 1987; because of the planned project closeout, no changes to the SRP-RD have been accepted by the SRP Change Control Board. However, several changes in the document had been identified in response to review comments from DOE and from project participants, and as a result of the usage of the document as a requirements document for the Exploratory Shaft Facility design.

Concurrent with the work to develop a project requirements document, work proceeded on the development of a system design description. The version corresponding to SRP-RD, Revision 4, was SRPO's Salt Repository Project Mined

Geologic Disposal System Description Document (SRP-MGDS-DD) (1987b). The SRP-MGDS-DD provides a detailed summary reference description of the SRP-MGDS, including the subsystems for site, waste package, and repository, and the interfaces with the accessible environment, society, waste transportation, and waste sources. The SRP-MGDS-DD is a controlled document in the SRP baseline identified as SRP/B-19.

1.2.5 Baseline Management System

To aid in project control, the SRP established cost, schedule, and technical baselines, and the project activities were managed against those baselines. In the SRP baseline management system, the cost and schedule baselines were the responsibility of WBS 9.0 - Project Management and are discussed in Chapter 9 of this report; the technical baseline was the responsibility of WBS 1.0 - Systems.

The SRP technical baseline was to have included any documents, data, drawings, etc. that supported the technical development of the MGDS by providing definitions, descriptions, requirements, and criteria of the various MGDS systems, subsystems, and interfaces. Changes to the technical baseline were controlled so that the baseline was technically correct and the potential impact of changes on other systems and subsystems were analyzed and accepted. The dissemination of the technical baseline to project participants was accomplished in a controlled fashion.

The material in the SRP technical baseline, at project closeout, included requirements, descriptions, and technical data. The requirements material included the SRP Requirements Document and material adopted from the OCRWM and OGR baselines. The description material included the SRP MGDS Description Document and material adopted from the OCRWM and OGR baselines. The technical data in the baseline included the SRP Technical Data Base and the SRP Information Sheets; a discussion of these items is provided in Section 1.3, Technical Data Base Management, of this report.

The procedures for the operation of the SRP baseline management system are given in SRPO's SRP Baseline Procedures Notebook, SRP/B-1 (1985). This document provides a description of the baseline management concept, establishes the SRP baseline itself, and provides procedures to be followed for controlling changes to that baseline.

It should be noted that the term "baseline management," as used in the project, includes configuration management. However, in response to a DOE requirement, a separate configuration management plan was developed. This plan is given in an unpublished SRPO document entitled "SRP Configuration Management Plan" (SRPO, 1987a). It had been planned to incorporate information from the configuration management plan into a revision of the baseline procedures notebook.

One aspect of baseline or configuration management that merits special attention is computer code control. The use of computer codes for the design, analysis, and performance assessment of the mined geologic disposal system requires conformance with the NRC guidelines given in NUREG-0856, "Final Technical Position on Documentation of Computer Codes for High-Level Waste

Management" (NRC, 1983). The project implementation of NUREG-0856 was through the use of the following BPMD Technical Management Procedures (BPMD, 1987):

- TMP-12, "Verification and Validation of Engineering/Scientific Computer Codes"
- TMP-21, "Computer Code Transfer"
- TMP-25, "Change Control of Engineering/Scientific Computer Codes"
- TMP-31, "Requirements of Engineering/Scientific Computer Codes."

It had been planned to incorporate elements of the above procedures into a revision of the SRP Baseline Procedures Notebook to better address computer code configuration management.

1.2.6 SRP Transportation Program

The SRP Transportation Program was broadly concerned with the transportation system required to support site characterization and the repository construction, operation, and decommissioning and closure. This system included both nuclear waste transportation and nonnuclear transportation activities. The primary responsibilities of WBS 1.0 - Systems were for program coordination, integration, and direction.

The early transportation activities within the project focused on defining the transportation system and assessing its potential impacts. This work supported the transportation-related information provided in the environmental assessments for the salt sites. The results of this work are provided in the published environmental assessments, and for the project the most relevant is the Environmental Assessment, Deaf Smith County Site, Texas, DOE/RW-0069, May 1986.

A key element in the SRP transportation program was the "Salt Repository Project Transportation Program Plan," an unpublished SRPO document (1987g). The document addresses the organization and implementation of the SRP transportation and transportation-related activities across all involved work breakdown structure elements. It describes the process of documentation and activity development from site characterization to license application. It identifies the major activities of the program:

- Monitoring
- Access corridor identification
- Alternative route identification
- Alternative route identification for salt disposal
- Emergency-response preparedness
- Development of requirements for transportation system interfaces.

The project work in each of the above activities is summarized below.

The monitoring activity was concerned with identifying the impacts of the transportation system on the local area, especially during site characterization. The monitoring studies were to be detailed in a transportation site study plan; the plan was developed but not finalized for the SCP.

The access corridor identification activity was concerned with determining the optimum road and railroad links from the Deaf Smith County site to the major highway and railway facilities in the site vicinity. The major activity in this area was to develop a methodology for access corridor selection. It appeared that the best approach would be to use a Geographical Information System similar to that utilized in the NUS report Rail Transportation Corridor Analysis Report: Deaf Smith County Location in the Palo Duro Basin, Texas, BMI/ONWI-617, October 1986, but with some modifications to accommodate the techniques utilized in "Estimated Transportation Routes to a Candidate Salt Repository Site in Deaf Smith County, Texas," an unpublished report by Joy, D. S., and Johnson, P. E., of Oak Ridge National Laboratory (ORNL), dated October 1987.

The alternative route identification work was to determine road and railway routing from the Texas border and in-state reactors to the repository access corridors. Work in this area is summarized in the ORNL report by Joy and Johnson cited above.

The alternative route identification for salt disposal related to the transportation issues associated with disposing of the excess salt from the repository. Work in this area is addressed in "An Approach to Selecting Routes Over Which to Transport Excess Salt From the Deaf Smith County Site," an unpublished report by Battelle's Office of Transportation Systems and Planning (OTSP), dated September 1987.

No significant work on the task concerning emergency-response preparedness was accomplished.

The development of requirements for transportation system interfaces was concerned with the interfaces between the transportation system and the mined geologic disposal system, the national transportation system, the accessible environment, and the societal system.

The interface between the transportation system and the mined geologic disposal system was addressed primarily in terms of the interface for nuclear waste activities in the following unpublished reports: "Operational Analysis Supporting the Definition of the Repository/Transportation Interface" by Peterson, R. W., et al., dated October 1985 and "Salt Repository Project Transportation System Interface Requirements: Transportation System/Repository Receiving Facility Interface Requirements," by Smith, L. A., et al., dated January 1988. Additional information is given in the SRP Closeout Task Report by ONWI entitled "Waste Handling Building Design Tradeoff Studies" (ONWI, 1988b).

1.3 TECHNICAL DATA BASE MANAGEMENT (WBS 1.3)

Technical data base management involved the establishment and maintenance of a base of significant technical data that would be used in evaluating the performance of the waste isolation system. This included activities to ensure that data could be documented, traced, and controlled.

One WBS-1.0 responsibility in the technical data base area related to the computerized SRP Technical Data Base (SRP/TDB). The SRP/TDB was a component of the Technical Data Management System (TDMS), which itself was part of the SRP Integrated Data Management System (IDMS). Information on the TDMS and the IDMS can be found in the published document by B. A. Rawles entitled Salt Repository Project Information Systems Description, BMI/ONWI-635, July 1987, and in Chapter 9, Project Management, of this report.

The other WBS-1.0 responsibility related to SRP Information Sheets. The SRP Information Sheets were developed as an efficient mechanism for baselining technical data and were extensively used to support the preparation of the SRP Site Characterization Plan.

1.3.1 Technical Data Base System

The computerized SRP Technical Data Base was a system for collecting, organizing, and managing technical data. It provided online access to summary numerical data and appropriate data qualifiers. It supported the generation of a technical data handbook and user-defined ad hoc reports. The data base resided on a VAX computer. It initially utilized a data base management software package known as BASIS, but was being converted to a relational data base management system developed by Battelle known as DM.

The SRP/TDB contained over 500 records covering 71 topics organized into nine distinct areas. These areas included geology, hydrology, background environment, background socioeconomics, environmental impacts, material properties, exploratory shaft design, source terms, and waste package. Documentation of the SRP/TDB is provided in several unpublished documents including a users manual, an operational manual, a functional specification, and a system design requirements document.

Toward the end of the project, consideration was being given to major redirection of the SRP/TDB. The concept being considered involved using the SRP Information Sheets directly in a data base, with data and information extraction based on a text processing-type data base management scheme.

1.3.2 SRP Information Sheets

The SRP Information Sheets were used to record geotechnical, environmental, socioeconomic, and other site reference data parameters and values for baselining. They were also used to record engineering and technical design parameters that required baselining.

The SRP Information Sheets were organized into five volumes:

- Synthetic Geotechnical Design Reference Data for the Deaf Smith Site, SRP/B-11 (SRPO, 1986a)
- Salt Repository Project Reference Data for the Deaf Smith Site - Environmental, SRP/B-16 (SRPO, 1987c)
- "Salt Repository Project Reference Data for the Deaf Smith Site - Geotechnical" (SRPO, 1987d)
- "Salt Repository Project Reference Data for the Deaf Smith Site - Repository"
- "Salt Repository Project Reference Data for the Deaf Smith Site - Waste Package" (SRPO, 1987e).

The SRP Information Sheets in the Synthetic Geotechnical volume provided information pertaining to hydrogeology, rock, soil, seismic, and stratigraphy areas. The volume contained over 70 Information Sheets. This volume would have been rescinded when complete information was available in the Geotechnical volume.

The set of SRP Information Sheets included in the Environmental volume addressed surface water hydrology, climatology and meteorology, and site reclamation. The volume included over 40 Information Sheets.

The Geotechnical volume contained over 50 Information Sheets. This volume provided information pertaining to geology, geoengineering, hydrology, and geochemistry.

The Repository volume was never compiled. It was intended to house SRP Information Sheets for the repository that addressed important repository design results from the SCP-CDR and important design parameters for the Advanced Conceptual Design.

The SRP Information Sheets in the Waste Package volume provided information pertaining to the waste package including waste form, container, canister, and waste package near-field environment. Over 40 Information Sheets are contained in the volume.

As mentioned earlier, the development of SRP Information Sheets was used to facilitate the development of the SRP SCP and to ensure that the SCP reflected information and data in the project technical baseline. For information on the final status of the SRP Information Sheets the SRP Closeout Task Report by ONWI entitled "SCP Closeout" (Guzzetta, 1988) should be consulted.

1.4 TOTAL SYSTEM PERFORMANCE ASSESSMENT (WBS 1.4)

Total system performance assessment is the analysis of the mined geologic disposal system (MGDS), considering its comprising subsystems (site, waste package, and repository) to determine its ability to satisfy waste isolation and nuclear safety-related criteria. This analysis includes the integration of the MGDS subsystems and the interactions of the subsystems with each other, during both the preclosure and the postclosure phases. The material below addresses the project activities in performance assessment by focusing on the following:

- Development of performance assessment concepts
- Preclosure performance assessment developments
- Postclosure performance assessment developments
- Sensitivity and uncertainty analysis developments
- Performance allocation.

The project's work in the area of performance assessment was performed by the project staff and two major subcontractors: Pacific Northwest Laboratory (PNL) and INTERA Technologies, Inc. The PNL work was initially performed under the Waste Isolation Safety Assessment Program (WISAP), and later under the Assessment of Effectiveness of Geologic Isolation Systems (AEGIS) Program. The INTERA work was performed under the Systematic, Comprehensive Evaluation of Performance and Total Effectiveness of Repositories (SCEPTER) Program.

1.4.1 Development of Performance Assessment Concepts

In the early phase of the project, performance assessment was mainly concerned with the postclosure phase of the repository. The focus was on defining the components of a waste isolation system, determining the ways in which the system could fail or the isolation capability could be degraded or compromised, and developing methodologies for analyzing the performance of the system which could handle both events and processes. This early work is best summarized in two published reports by Burkholder, H. C., Waste Isolation Performance Assessment - A Status Report, ONWI-60, November 1979, and The Development of Release Scenarios for Geologic Nuclear Waste Repositories: Where Have We Been? Where Should We Be Going?, ONWI-163, September 1980. It should be noted that during this early phase, the project performance assessment work was not yet focused on a repository in a salt medium.

During the next phase, the project was focused on a salt repository, although no specific site had yet been selected. At this time, the influence of a salt medium on the technical tools needed for performance assessment became clearer. A review of the project performance assessment work at this phase is provided by Brandstetter, A., and Gupta, S. K., "Salt Repository System Performance Assessment Capabilities," in DOE/NWTS-30, 1982, and Kircher, J. F., and Gupta, S. K., "Salt Site Performance Assessment Activities," in CONF-831217, 1983.

As the project's conception of performance assessment became clearer, the need for planning and control of the performance assessment work became stronger. This first effort at planning appeared in an unpublished DOE report "NWTS Performance Assessment Plan," DOE/NWTS-2, which went through several draft stages during the early 1980s. The first major planning document to be published was ONWI's Performance Assessment Plans and Methods for the Salt Repository Project, BMI/ONWI-545, dated August 1984. An updated discussion of the Salt Repository Project performance assessment plans and methods is presented in Section 8.3.5 of the SCP (DOE, 1988).

The project conducted performance assessments for the salt-site Environmental Assessments and further developed its performance assessment capabilities for the Deaf Smith County, Texas, site. The project's final conception of performance assessment is reflected in DOE's Salt Repository Project Site Characterization Plan (DOE, 1988), which discusses such supporting documentation as the program plan for performance assessment (SCP Section 8.3.5) and its six analysis plans: preclosure (SCP Section 8.3.5.1), repository (SCP Section 8.3.5.2), seals and shafts (SCP Section 8.3.5.2), site (SCP Section 8.3.5.2), total system (SCP Section 8.3.5.2), and waste package (SCP Section 8.3.5.2).

1.4.2 Preclosure Performance Assessment Developments

Preclosure performance assessment is concerned with the safety-related performance of the mined geologic disposal system during the period prior to its closure. The assessment thus focuses primarily on the operation of the repository, but includes retrieval, if required, and decommissioning and closure operations.

The early work in the performance assessment area involved reviewing past work on nuclear facility safety assessments (for example, Ensminger, D. A., et al., A Review of Safety Assessments of Nuclear Waste Management, ONWI-126, November 1980) and developing accident scenarios (see Yook, H. R., et al., Repository Preclosure Accident Scenarios, BMI/ONWI-551, September 1984).

The work led to the preclosure repository safety analysis type information provided in the environmental assessments for the salt sites, such as the Environmental Assessment, Deaf Smith County Site, Texas, DOE/RW-0069, May 1986. The supporting calculations for the EA are provided in Preclosure Radiological Calculations to Support Salt Site Evaluations, BMI/ONWI-541 (Waite, 1984) and BMI/ONWI-541 (Rev. 1) (Waite et al, 1986).

Subsequent work in the preclosure area was closely tied to the repository design and analysis being performed by Fluor, the architect-engineer for the salt repository. This work is documented in the unpublished report by Waite, D. A., et al., "Preclosure Performance Assessment Review of SCP-CDR," October 1987. The plans for future work in this area include the "Preclosure Performance Assessment Analysis Plan" and the "Repository Performance Assessment Analysis Plan," developed by ONWI.

Fundamentally, the preclosure performance assessment of the repository is similar to the safety analysis or risk assessment of other nuclear facilities. Accordingly, many analysis tools were available for application to the repository assessment and significant computer code development was not required.

1.4.3 Postclosure Performance Assessment Developments

The postclosure performance assessment of the mined geologic disposal system is concerned with the effectiveness of the waste isolation over a period of time reaching thousands of years into the future. The assessment thus focuses on the effects of the natural environment on the repository, the effects of the repository on the natural environment, climatic and geological changes that might occur over such a long period, and potential human activities that could jeopardize the effectiveness of the waste isolation.

The early phase of postclosure performance assessment work by the project was briefly discussed earlier. The later work, in support of the salt site environmental assessments, is addressed in Section 6.4, Performance Assessments, of the Environmental Assessment, Deaf Smith County Site, Texas, DOE/RW-0069, May 1986.

Subsequent project work was focused on the postclosure performance assessment aspects of the SRP Site Characterization Plan. This work is reported in ONWI's "Postclosure Performance Assessment Review of the SCP-CDR", (1987b) and in the Performance Assessment Program Plan and the Analysis Plans for the Repository, Seals, Site, Total System, and Waste Package.

One aspect of the postclosure performance assessment of a waste isolation system that was given special attention was the need for computer codes to model the performance of the system. Some of the more significant computer codes that were under development by the project are identified below; for an update of the final status of these codes, see the discussion on performance assessment code documentation contained in Section 8.3.5 of the SCP (DOE, 1988).

- BRINEMIG. The BRINEMIG code calculates the rates and quantities of brine migration to a waste package emplaced in salt. The code is an explicit time-marching finite-difference code which solves a mass balance equation and uses the Jenks equation to predict velocities of brine migration. The model documentation and the detailed results of the model used for the salt sites Environmental Assessments are given in Expected Brine Movement at Potential Nuclear Waste Repository Salt Sites, BMI/ONWI-564, August 1987, by McCauley, V. S., and G. E. Raines.

- CFEST. The CFEST code is a coupled fluid, energy, and solute, three-dimensional transport code that simulates seasonal energy storage in confined aquifers. It treats single-phase Darcy flows in a horizontal or vertical plane, or in fully three-dimensional space, under nonisothermal conditions. Both steady-state and transient simulations are possible. An early version of the code is documented in A Multidimensional Finite-Element Code for the Analysis of Coupled Fluid, Energy, and Solute Transport (CFEST), PNL-4260, 1982, by S. K. Gupta et al. The current version of CFEST is documented in Coupled Fluid, Energy, and Solute Transport (CFEST) Model: Formulation and User's Manual, BMI/ONWI-660, October 1987, by S. K. Gupta et al.

- CFEST-BRINEMIG. The CFEST-BRINEMIG code is a modification to the CFEST code to provide an option for convective fluid flow according to the Jenks type equation. The changes required to the CFEST code and documentation for brine migration according to the Jenks equation are addressed in an unpublished memorandum by McNulty, E. G., entitled "CFEST-BRINEMIG Controlled Version of Version 1.0," dated September 1987.

- CFEST-INV. The CFEST-INV code is a stochastic hydrology code that augments the CFEST code in the areas of data processing, model calibration, performance prediction, error propagation, and data collection guidance. The code uses kriging, finite element modeling, adjoint sensitivity, statistical inverse, first-order variance, and Monte Carlo techniques to develop performance-driven data collection schemes and to determine the waste isolation capabilities, including uncertainties, of waste repositories. The code is documented in an unpublished report by J. L. Devary entitled "The CFEST-INV Stochastic Hydrology Code: Mathematical Formulation, Application, and User's Manual" (1987a), and the usage of the code is illustrated in an unpublished report by J. L. Devary entitled "Evaluation of Groundwater Travel Times in the Wolfcamp Formation Using the CFEST-INV Stochastic Hydrology Library" (1987b).

- PTRACK. The PTRACK code tracks the path of a radionuclide particle released from a nuclear waste repository into a ground-water flow system in a two-dimensional representation of a stratified geologic medium. The code calculates the time required for the particle to travel from the release point to the specified boundary. The physical properties of the geologic setting and the ground-water flow system can be treated as fixed values or as random variables sampled from their respective probability distributions. The model is documented in an unpublished report by INTERA entitled "PTRACK - A Particle Tracking Program for Evaluating Travel Path/Travel Time Uncertainties" (1986a). The model was used to provide results for the Environmental Assessments, and detailed results are documented in an unpublished report by INTERA entitled "Travel Path/Travel Time Uncertainties of Salt Sites Proposed for High-Level Waste Repositories" (1986c).

- SALT-GSM. The SALT-GSM code is a Monte Carlo computer simulation of the important geologic events related to a nuclear waste repository in a salt medium. Processes taken into account include climate change, tectonics, salt dissolution, and geomorphic events. The code is based on the Geologic Simulation Model developed at PNL, but includes only important features associated with Palo Duro salt sites. The code is documented in a report to be published by PNL entitled "Salt Repository Project: Preliminary Palo Duro Geologic System Model (PDGSM)," by Petrie, G. M., dated September 1987.

- SCATTER. The computer code SCATTER simulates several processes leading to the release of radionuclides to the site subsystem and then simulates transport via the ground water of the released radionuclides to the biosphere. The processes accounted for to quantify release rates to a ground-water migration path include radioactive decay and production, leaching, solubilities, and the mixing of particles with incoming uncontaminated fluid. Results are cast as radionuclide discharge rates and total discharge to the accessible environment. The code is documented in an unpublished report by INTERA entitled

"SCATTER: Source and Transport of Emplaced Radionuclides - Code Documentation," March 1987.

- SYSNET. The SYSNET code is a systems network model for assessing potentially disruptive scenarios of geologic repositories in a salt medium. The model features a general three-dimensional network topology and simulates the processes of flow, heat transport in rock, heat transport in fluid, brine transport, salt creep, dissolution, and precipitation. The code uses relatively simple semianalytic algorithms so that it may be implemented statistically to calculate distributions of various performance measures and sensitivities of performance measures to uncertain parameters. The model is documented in an unpublished report by INTERA entitled "SYSNET: A Salt-Site Systems Network Model for Scenario Assessments" (1986b).

- EQ3/EQ6. The EQ3 and EQ6 codes describe geochemical processes that influence the performance of nuclear waste repositories. EQ3 simulates a distribution of species and the resulting fluid is used in EQ6 to model a reaction path. EQ3/EQ6 can be applied to near-field and far-field performance assessment, to evaluate data acquisition needs, and to assist in data interpretation. The codes are described in a published report by INTERA entitled EQ3/EQ6: A Geochemical Speciation and Reaction Path Code Package Suitable for Nuclear Waste Performance Assessment, ONWI-472, May 1983.

- WAPPA. The WAPPA code is a barrier degradation code for evaluating the performance of a waste package in a geologic repository. The code treats the waste package as an axially symmetric system of coupled physical and retardation barriers, subject to several simultaneous degradation processes. The code consists of five process models that treat radiation, thermal, mechanical, corrosion, and leaching processes. The principal code outputs are the state of integrity of the waste package components as a function of time and the heat and radionuclide fluxes output from the waste package. The code is documented in a published report by INTERA entitled WAPPA: A Waste Package Performance Assessment Code, ONWI-452, April 1983, and in Verification of the 2.00 WAPPA-B Code, BMI/ONWI-653, July 1987, by Tylock, B., et al. The WAPPA code was used to provide results for the Environmental Assessments, and detailed results are documented in Expected Waste Package Performance for Nuclear Waste Repositories in Three Salt Formations, BMI/ONWI-655, August 1987, by G. Jansen.

- SPECTROM-41. The SPECTROM-41 code is a finite element program for the analysis of two-dimensional, axisymmetric conductive heat transfer. The code is part of the SPECTROM (Special Purpose Engineering Codes for Thermal/Rock Mechanics) series of computer programs developed to address the unique problems resulting from the storage of radioactive waste in a geologic repository. The code is documented in User's Manual for SPECTROM-41: A Finite-Element Heat Transfer Program, ONWI-326, June 1983, by Svalstad, D. K., and in an unpublished report by Svalstad of RE/SPEC Inc. entitled "Documentation of SPECTROM-41: A Finite-Element Heat Transfer Analysis Program," dated June 1985.

- TEMP. The TEMP code calculates temperatures in a geologic repository for nuclear waste. It will calculate the incremental temperature contributed by a single heat source, by an infinite array of heat sources, or by heat sources geometrically arranged in a finite array. The code uses a semianalytical technique for solving the equation for a heat-producing finite-length line source in an infinite isotropic medium. TEMP uses a temperature-dependent thermal conductivity. Temperature contributions from individual heat sources are superimposed using a special transformation technique to determine the temperature at a specific location and time for materials with temperature-dependent properties. The model is documented in TEMP: A Finite Line Heat Transfer Code for Geologic Repositories for Nuclear Waste, BMI/ONWI-668, October 1987, by Wurm, K. J., et al. The model was used to provide results for the Enviromental Assessments, and detailed results are documented in Expected Near-Field Thermal Performance for Nuclear Waste Repositories at Potential Salt Sites, BMI/ONWI-652, August 1987, by E. G. McNulty.

- SPECTROM-32. The SPECTROM-32 code is a finite element program for the analysis of two-dimensional, axisymmetric nonlinear structural problems including those of plasticity and creep. Several special features of the code are body forces, sliding interfaces, multiple materials, excavation, thermally dependent properties, nonlinear strain hardening, nonassociative flow rules, and automatic time stepping. The code is part of the SPECTROM (Special Purpose Engineering Codes for Thermal/Rock Mechanics) series of computer programs developed to address the unique problems resulting from the storage of radioactive waste in a geologic repository. The code is documented in an unpublished report by Callahan, G. D., et al. of RE/SPEC Inc. entitled "Documentation of SPECTROM-32: A Finite Element Thermomechanical Stress Analysis Program," 1987.

- GRESS. The Gradient Enhanced Software System, GRESS, is a general software package that can be used as a tool to facilitate sensitivity analyses of computer codes. It is a pre-compiler that uses computer calculus to obtain sensitivity coefficients by differentiation of arithmetic statements. The project was helping support the development of GRESS because of its potential application to sensitivity and uncertainty analysis. GRESS is documented in GRESS, Gradient Enhanced Software System, Version D, User's Guide, ORNL/TM-9658, 1985, by E. M. Oblow. An application of GRESS is given by Pin, F. G., et al. in a paper entitled "An Automated Sensitivity Analysis Procedure for the Performance Assessment of Nuclear Waste Isolation Systems," 1986.

1.4.4 Sensitivity and Uncertainty Analysis Development

Sensitivity and uncertainty analyses of the computer codes used to model the performance of the waste isolation system were developed by the project to assess the degree of confidence one should have in the model predictions. In addition, these analyses were used to establish requirements for measuring or determining parameters that affect the performance of the isolation system.

Early project work in the area of sensitivity and uncertainty analyses is given in the following published reports: Harper, W. V., et al., "Evaluation of Sensitivity/Uncertainty Analysis Techniques for Salt Repository Performance Assessments," in DOE/NWTS-30, 1982; Thomas, R. E., Uncertainty Analysis, ONWI-380, March 1982; Harper, W. V., Sensitivity/Uncertainty Analysis Techniques for Nonstochastic Computer Codes, ONWI-444, May 1983; and A Proposed Approach to Uncertainty Analysis, ONWI-488, July 1983, by INTERA. The work is also reported in two conferences/workshops documented in NRC's Proceedings of the Symposium on Uncertainties Associated with the Regulation of the Geologic Disposal of High-Level Radioactive Waste, NUREG/CP-0022, March 1982, and INTERA's Workshop on Uncertainty Analysis of Postclosure Nuclear Waste Isolation System Performance, ONWI-419 (1983e).

As explained in the above reports, the project was developing two sensitivity/uncertainty analysis approaches: statistical methods and the deterministic approach. A very useful comparison of these approaches is provided in Harper, W. V., and Gupta, S. K., Sensitivity/Uncertainty Analysis of a Borehole Scenario Comparing Latin Hypercube Sampling and Deterministic Sensitivity Approaches, BMI/ONWI-516, October 1983.

The use of the statistical method for sensitivity/uncertainty analysis is based on concepts developed by R. L. Iman of Sandia National Laboratory. The project's application of this method is given in "Travel Path/Travel Time Uncertainties at Salt Sites Proposed for High-Level Waste Repositories," an unpublished report by INTERA (1986c).

The use of the deterministic method for sensitivity/uncertainty analysis is based on concepts developed by E. M. Oblow of Oak Ridge National Laboratory. This method uses the computer code GRESS, Gradient Enhanced Software System, to obtain sensitivity coefficients for the parameters in the computer code being analyzed. The project's application of the deterministic method to sensitivity/uncertainty analysis is given in the published INTERA report Adjoint Sensitivity Theory for Steady-State Ground-Water Flow, BMI/ONWI-515 (1983a).

1.4.5 Performance Allocation

Performance allocation was a procedure used by the project to allocate performance objectives to subsystems of the mined geologic disposal system (MGDS). Performance allocation was implemented during the development of the site characterization plan to provide a rationale for subsystem requirements and site characterization test plans.

The performance allocation procedure used by the project was comprised of the following nine steps.

1. Identify regulatory and other performance objectives.

2. Disaggregate MGDS structure (Figure I-3).

3. Allocate performance objectives to structural units.

4. Assign functions to structural units and signify importance.

5. Identify performance measures and parameters for each structural unit and its assigned functions.

6. Identify required or estimated values and existing values of each performance measure and parameter.

7. Provide indications of confidence.

8. Describe the data gap.

9. Derive the test plans.

This procedure was applied to six basic performance objectives: radiation protection, retrievability, containment, release rate, ground-water travel time, and total system release. Each of these basic performance objectives was further specified in terms of lower level objectives; for example, the radiation protection objective included limiting radiation levels, occupational dose, and airborne radioactivity. Thus, the performance allocation process was performed for about 30 specific regulatory-related performance objectives.

The project work in performance allocation is documented in an unpublished ONWI report entitled "Preliminary Performance Allocation Report" (1987c). Additional information can be found in DOE's Salt Repository Project Site Characterization Plan.

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CHAPTER 2

WASTE PACKAGE (WBS 2.0)

The concepts for the waste package and its components, materials, and functions have undergone significant changes since the beginning of the Salt Repository Project. At the outset, the wastes identified for geologic disposal were solidified high-level waste from reprocessing commercial spent fuel (CHLW), solidified high-level waste from defense activities (DHLW), and transuranic contaminated wastes (TRU) primarily from commercial fuel reprocessing. The solidification process for these wastes had not been selected. With the deferral of commercial fuel reprocessing by the Carter Administration, the primary waste form for geologic disposal became spent fuel from commercial power reactors. CHLW was limited to the small quantity then in storage at West Valley, New York, which more closely resembled DHLW than the more intensely radioactive commercial CHLW which had been expected. Without commercial reprocessing, little TRU waste would be generated also. The leading solidification process for high-level waste (HLW) was the vitrification of the calcined waste by in-can melting with a borosilicate glass frit. Later, this was replaced by injecting the liquid waste along with the glass frit into a continuous electric melter and casting the molten glass in a canister.

The waste package was considered the waste form, either solidified HLW or spent fuel, in a canister. The canister was to provide containment of the waste during handling, transporting, and emplacing the waste in the repository. The canister had no postemplacement function; the isolation of the emplaced waste was the function of the geologic media. However, the early drafts of the NRC regulations and the EPA standards showed that the regulators were considering requiring the waste package to (1) provide containment through the period when temperatures were elevated by the heat from the wastes and (2) assure low release from the engineered systems thereafter. This led to developing long-life containers (initially called overpacks) in which to seal the canistered waste, and investigating packing materials to provide additional impedance to the release of radionuclides from the waste package.

Conceptual waste package design studies were begun about 1980 with tentative design requirements from the DOE since the NRC regulations had not been finalized, and with assumptions for the environmental conditions which the waste packages would experience in the repository since no site had been selected for a repository.

When finalized in 1983, two of the NRC requirements had a large influence on the conceptual design of the waste package. The first was that the waste package must provide substantially complete containment of the waste for 300 to 1,000 years after closure of the repository. The second was that after the containment period to 10,000 years after repository closure, the release of each radionuclide from the engineered barrier system (assumed by DOE to be the outer surface of the waste package) must be a small fraction ($<1x10^{-5}$) of that calculated to be present 1,000 years after repository closure.

It was assumed that the repository would be located in a deposit of high-purity salt with very low water (brine) content. The water available to react with the waste package components was believed to be from brine that migrated up a thermal gradient to the waste package, thus the brine would be of limited quantity and would cease to move to the waste packages as the temperatures equilibrated and the gradients disappeared. This small quantity of water could be consumed by reacting with a small fraction of the container so the corrosion of the container could be expected to cease and no water would be available to react with the waste form.

Parametric studies were conducted to evaluate the waste package temperatures as a function of heat output of the waste and areal heat loading of the repository. The results of these studies showed that, with the higher thermal conductivity of salt than most other rocks being considered for repositories, higher heat loadings could be accommodated in a waste package and meet temperature limits placed on the components of the waste package and the host rock.

Materials research studies were conducted to select materials for waste forms, long-lived containers, and packing materials. These studies emphasized testing materials under a range of expected service conditions with the objective of developing a basic understanding of the phenomena controlling the performance of the materials to allow predicting the long-term performance to the degree necessary to show compliance with the regulations. A part of each of these studies was the development of phenomenological models based upon fundamental principles for incorporation in system performance codes for assessment of the performance of the waste package and the total repository system.

Low-carbon steel was selected as the reference container material for the waste package. The rate of general corrosion of the steel in sodium chloride brines was sufficiently low that a small thickness of steel (about 1 inch) was provided as a corrosion allowance so the structural thickness of the container could be protected through the containment period. The low-carbon steel was not believed to be susceptible to localized forms of attack under the expected service conditions. The corrosion allowance was expected to consume the available water in the brine so no water would be available to dissolve or transport radionuclides from the waste form if the container were penetrated.

The Deaf Smith County site was identified in May 1986 for site characterization. Limited data from bore holes near the site showed that the host rock was largely halite but was intersected by clay interbeds. The average water content of the host rock was estimated to be about 4 volume percent brine. The brine content was higher than previously assumed and, if the interbeds contained interconnected porosity, they could allow delivery of brine from greater distances than had been assumed. Thus, it could no longer be assumed that the quantity of water would be limited so container corrosion would cease and no water would react with the waste form. Also, the analyses of brines from inclusions in the halite from cores taken in the region showed high concentrations of magnesium (>50,000 ppm). The potential for high magnesium brines which could cause more rapid corrosion of the container necessitated the consideration of alternative materials to low-carbon steel for the container. Also, the potential for larger volumes of brine which could react with the waste form made it necessary to consider a change in the strategy for showing compliance with the gradual release requirement for the waste package by allocating performance to other components of the waste package. As a result, the plans for studies of packing materials were renewed.

This chapter presents a summary of the waste package environment which includes the ambient (preemplacement) thermal, in situ stress, brine volume, and brine chemistry conditions expected in the host rock at the emplacement horizon. A section on waste package design gives a summary description of the waste forms for disposal, the design requirements, the reference conceptual design, the design assessment, the licensing strategy, alternative designs, and pre-advanced conceptual design studies. A third section summarizes the studies that have been conducted to evaluate materials for the waste package components. The last section briefly describes modeling studies that have been conducted for the performance assessment for the waste package. Each of these sections, environment, design, component material studies, and performance assessment, constitute subelements of the waste package work breakdown structure.

2.1 WASTE PACKAGE ENVIRONMENT

This section summarizes the expected ambient conditions and the response of the host rock temperatures, in situ stress, brine migration, and brine chemistry to the excavation and emplacement activities in the repository. This summary is extracted from Sections 7.1 and 7.4 of the Site Characterization Plan (SCP) (DOE, 1988).

2.1.1 Host Rock Temperatures

The temperature of the host rock prior to excavation and waste emplacement (i.e., the ambient condition) is expected to be approximately 27°C based upon water pumping data from the region (DOE, 1986, pp. 3-104). The heat from the emplaced wastes will increase the rock temperature depending upon the proximity of the rock to the waste packages and the amount of heat emanating from the waste packages. As the heat from the waste packages declines by radioactive decay of the waste, the rock and waste package temperatures will decline. The temperatures of the waste package components and the host rock will have a large influence on the performance of the waste package and repository. The expected postemplacement temperatures of the waste packages and the host rock are dependent on the repository and waste package design and are discussed in Section 2.2.4.1.

2.1.2 In Situ Stress

Thermomechanical analyses were performed for the host rock adjacent to the waste package to estimate the peak radial stress as a function of areal thermal loading, size of gap between the wall of the emplacement hole and the container, thermal conductivity of the salt, and creep rate of the salt (Fluor, 1985). Modeling of the container region is not straightforward and requires simplifying assumptions. The results of the analysis showed that the closure of the emplacement hole around the waste container is rapid in the beginning, but the closure rate decreases with time and tends to reach a steady-state closure rate. With a minimum size gap, the peak radial stress exerted on the container does not exceed the in situ lithostatic stress which is estimated to be 18 MPa.

2.1.3 Brine Migration

It is expected that the excavation of the entries and emplacement holes will significantly perturb the ambient system. Specifically, it is known that brine that is present in bedded rock, but immobile under ambient geologic conditions, can be mobilized by the waste package temperature field and/or by the pore-pressure gradients arising from the localized relief of the in situ stress during excavation, salt creep, and differential thermal expansion (Shefelbine, 1982; Sherer, 1987). The waters normally found in bedded salt deposits are in the form of intracrystalline inclusions, intercrystalline brine, chemically bound water, and water associated with the interbed impurities. The temperatures near the waste packages are not expected to be high enough to mobilize the chemically bound water. However, temperatures will be high enough to cause evaporation and drying in the pores and microcracks that connect with the emplacement openings (Olander, 1982). The focus of attention is on the movement of intracrystalline and intercrystalline fluids in the polycrystalline rock salt.

It is known that liquid inclusions move up a temperature gradient in salt, and the theory for the rate of movement is fairly well developed. In addition, it is known that gas-liquid inclusions with a sufficiently high gas content move down the temperature gradient, and the theory for the rate of movement is fairly well developed (Jenks, 1979). The behavior of fluid inclusions is less clear when they reach grain boundaries. It has been recognized that there are other potential mechanisms for brine migration in rock salt. In general, the operative mechanisms are thought to involve the flow of fluids under the influence of pressure gradients through connected pores and microcracks (Shefelbine, 1982; Olander, 1982; Sherer, 1987; Nowak, 1986). The emplacement hole is expected to interrupt interbeds and stringers that may be laterally extensive and contain connected porosity. Potential sources of pressure gradients that could cause fluid flow in the connected pores and microcracks include differential thermal expansion between the pore fluids and the host rock, water vapor pressure, stress gradients in the host rock, hydrogen generation from container corrosion, and connections with the regional hydraulic system.

The details of the host rock structure and the mechanisms of fluid flow needed to confidently predict potential inflow and outflow of brine from the waste packages in the LSA4 were being developed. However, in the case where laterally extensive hydraulic connections within the regional hydraulic system are absent, there is not expected to be any pressure driven flow in the vicinity of the emplaced waste package. As the temperature begins to decline, the temperature gradients causing the thermal transport of brine would approach zero and the migration of intercrystalline brine would cease. Also, it is expected that the pressure gradients which might support brine inflow would be small or nonexistent after packing, consolidation, and closure of the emplacement hole annulus. Thus, within a few years the hydrology in the LSA4 would return to the ambient.

2.1.4 Brine Chemistry

As a consequence of the thermal and radiation fields within and around the waste package, as well as possible evaporation, condensation, contamination, and chemical reactions with the emplaced materials, the composition of the brine will differ from the ambient composition. To the extent that evaporation of ambient brine takes place, halite will precipitate and the concentrations of potassium and magnesium will increase, at least until saturation is reached in some potassium or magnesium salt. Organic contaminants arising from diesel fumes, soil, and refuse can be potentially serious because many such substances are capable of forming soluble complexes with radionuclides. The viability of microorganisms that promote corrosion in saturated salt solutions at elevated temperatures is unknown and should be investigated. The mass of air left in the emplacement hole and adjacent back-filled entry is small in comparison with the mass of the container. Thus, whereas the environment will initially be oxidizing owing to the presence of residual oxygen, it should soon become reducing. Similarly, the effect of reactions with residual nitrogen and carbon dioxide from the trapped air is expected to be negligible. Radiolysis effects outside of the container are not expected to significantly affect either the brine or the halite. The principal anticipated reaction of brine with the steel container is the production of various iron oxides and hydroxides together with hydrogen. The reaction may tend to raise the pH. This will compete with the hydrolysis of magnesium ions in the brine which will tend to reduce the pH, and the net effect of the two reactions on the pH is uncertain. Also, it is presently unknown whether the rate of hydrogen generation as compared to its rate of diffusion away from the waste package will lead to the production of a separate gas phase.

If brine contacts the waste form after container failure, alpha radiolysis of the brine can produce oxidizing species and may affect the pH. Reactions of the container material and the radionuclide species with oxidants and reductants is of special importance. The ambient brine will contain very low concentrations of species that participate in redox reactions. Thus, initially there is little to influence the redox state. Consequently, the waste package will exert the major control through reactions with steel and radiolysis. Conditions adjacent to the steel are expected to be strongly reducing. Diffusion of hydrogen from the package will maintain generally reducing conditions in the host rock. Adjacent to the waste form, brine is expected to be moderately to strongly oxidizing. Otherwise the chemical composition of the brine is expected to bear a close resemblance to the ambient brine with the addition of a moderate concentration of ferrous ions near the steel where conditions are reducing and of radionuclides dissolved from the waste form.

2.2 WASTE PACKAGE DESIGN

This section describes the waste forms planned for geologic disposal, the requirements for the waste package, the reference conceptual design for the waste package, and the materials being considered for the waste package components. Also, results are given of the thermal, structural, and radiation design analyses. The requirements, design, and analyses are described more fully by Westinghouse (1986) and ONWI (1988a). The licensing strategies and alternative designs if needed for licensing are discussed and preliminary activities to advanced conceptual design are described. The strategies are discussed more fully in Section 8.2 of the SCP (DOE, 1988) and in the waste package strategy document (ONWI, 1988b).

2.2.1 Waste Forms and Types

This section contains descriptions of the basic waste forms, solidified high-level waste from reprocessing spent fuel, spent fuel from commercial power reactors, and the types of these two waste forms included in the reference conceptual design for the waste package. The waste form will assist in controlling the release of radionuclides from the waste package during the post closure period of the mined geologic disposal system.

2.2.1.1 Solidified Waste From Reprocessed Fuel

Solidified high-level waste is produced from the liquid wastes containing all of the fission products and most of the transuranic products from the chemical separation of uranium and plutonium from spent reactor fuel. High-level wastes are generated as a result of the nuclear defense activities of the DOE in South Carolina, Washington, and Idaho and are presently in storage at those locations. No reprocessing of commercial spent fuel is being conducted in this country and none is planned in the near future. Some liquid high-level waste from commercial fuel reprocessing conducted in the past is stored at West Valley, New York, and will be solidified for disposal by the DOE.

The Savannah River Plant near Aiken, South Carolina, is expected to produce the first solidified high-level waste for disposal and the waste from this plant is the reference defense high-level waste for the waste package conceptual design. This waste, containing high-level waste sludge oxides, is a borosilicate glass which is poured, while molten, into a stainless steel canister 0.61 m in diameter and 3 m long. An optional sludge-supernate waste has higher heat output than the sludge-only waste but will be produced in smaller quantities. The maximum thermal output used for this waste form in the conceptual design study was 470 W per canister, however subsequently DOE specified a maximum of 800 W per canister. The stainless steel canister, supplied as part of the product, is suitable for handling and short-term surface storage of the waste. The design characteristics of this waste are given in Table 2-1.

West Valley high-level waste is the only commercial high-level waste expected to be received for disposal in the near future. West Valley waste is approximately the same size and weight, and has a similar heat load to defense high-level waste. Therefore, this waste has not received explicit analysis;

Table 2-1. Design Characteristics for Waste Types with Canisters (a)

Characteristic	Defense HLW	Spent Fuel			
		Intact		Consolidated	
		PWR	BWR	PWR	BWR
Canister Diameter (cm)	61	67.3	67.3	62	62
Canister Length (m)	3.0	4.24	4.24	4.00	4.35
Number of Fuel Assemblies	-	4	9	12	30
Net Waste Weight	1,470	1,840(b)	1,700(b)	5,530(b)	5,670(b)
Waste Form and Canister Weight (kg)	1,940	3,170	2,985	8,470	8,780
Maximum Power Output (W)	470	2,200(c)	1,620(c)	6,600(c)	5,400(c)

(a) ONWI, 1988a, Table 2-1.
(b) Expressed as kgU for Spent Fuel, MW/T for
(c) Heat output assumes 10-year decayed spent fuel, 33,000 MWD/T burnup for PWR fuel, and 27,000 MWD/T for BWR fuel.

and, although it is acceptable for disposal in the repository, West Valley waste is not considered separately from defense high-level waste in the conceptual design of the waste package.

2.2.1.2 Spent Fuel

Spent fuel will be received for disposal from commercial light-water nuclear power reactors. This waste form contains all of the fission products and transuranic radionuclides which were produced during service in the reactor which have not been removed by radioactive decay since removal from the reactor. With few exceptions, the spent fuel which has been removed from power reactors is being stored by the utilities in fuel storage pools at each reactor until it is accepted by the DOE for disposal or retrievable storage (proposed). During Phase 1 of the repository, intact spent fuel assemblies will be received from the utilities for disposal at the repository. The waste package conceptual design studies have shown that up to four pressurized-water reactor (PWR) assemblies or nine boiling-water reactor (BWR) assemblies can be accommodated in a waste package. The maximum heat output is 2.2 kw for the four-PWR package.

During Phase 2, most of the fuel assemblies will be disassembled and the fuel rods consolidated into a closely packed array to enable more fuel to be placed in a waste canister. Consolidated fuel rods from up to 12 PWR or 30 BWR assemblies with a maximum heat output of 6.6 kw can be placed into a canister of similar size to the intact fuel canister.

The design characteristics of intact spent fuel and consolidated spent fuel and their respective canister dimensions are given in Table 2-1.

2.2.1.3 Other Waste Types

Reactor-Consolidated Spent Fuel. Some utilities may consolidate some of their spent fuel to increase the capacity of their storage pools. It is assumed that reactor consolidated fuel will have the rods from two assemblies placed in a box with approximately the same external dimensions as an intact fuel assembly. When received at the repository, these boxes of consolidated fuel will be placed into a canister designed for intact fuel assemblies. With the rods from two assemblies in the space for one intact assembly, this configuration can have twice the heat output of the intact fuel package. This is less heat output than for the fuel rods in the consolidated spent fuel package so the reactor-consolidated fuel in the intact fuel package was not specifically analyzed during conceptual design.

Spent Fuel Hardware. Consolidating fuel at the repository during Phase 2 will result in a waste stream composed of spacer grids, end fittings, springs, etc. This waste will be radioactively contaminated externally with crud from the reactor and internally with activation products produced by exposure to neutrons in the reactor. However, the radioactivity content will be so low that it will produce very little heat so this will be considered a non-heat-producing waste. It is planned to dispose of this waste in the repository in containers designed for consolidated spent fuel. No analyses were conducted for the performance of this non-heat-producing waste since it will be disposed of in a container used for high-heat-producing waste.

2.2.2 Waste Package Requirements

The waste package must satisfy a number of requirements which are beyond the scope of this summary. However, the conceptual design has been most influenced by the Nuclear Regulatory Commission's two requirements in 10 CFR 60.113 which are (1) to provide substantially complete containment of the radionuclides for up to 1,000 years after repository closure and (2) to provide controlled release of radionuclides to a small fraction of their 1,000-year inventory for the period from 1,000 to 10,000 years after repository closure.

2.2.2.1 Substantially Complete Containment

The U.S. Nuclear Regulatory Commission (NRC) regulations for high-level nuclear waste disposal set performance objectives for the waste packages of the high-level waste (HLW) during the period after closure of the repository when the temperatures and radiation levels are highest. The performance objective for containment is as follows (10 CFR 60.113(a)(1)(ii)):

> ...the engineered barrier system shall be designed, assuming anticipated processes and events, so that: (A) Containment of HLW within the waste packages will be substantially complete for a period to be determined by the Commission taking into account the factors specified in 60.113(b) provided that such period shall be not less than 300 years nor more than 1,000 years after permanent closure of the geologic repository....

The NRC, in discussing its rationale for 10 CFR 60.113, has provided only a qualitative definition of substantially complete containment:

> It is expected that containment of the waste will be substantially complete, with releases during the containment time limited to a small fraction of the inventory present.

The Salt Repository Project (SRP) adopted the DOE interpretation and design objectives for the regulatory term "substantially complete containment." The interpretation in brief states: "The requirement would be met if a significant number of the waste packages were to provide total containment of the radioactivity within those waste packages or if the radioactivity released from the set of waste packages during the containment period were sufficiently small."

The following three design objectives were set as program goals:

1. By virtue of the intrinsic properties and design of the waste package components subjected to the range of conditions anticipated in the underground facility, 80 percent or more of the waste packages will retain all their radioactivity for a containment period of 1,000 years after permanent closure of the repository.

2. At any time during the containment period, at least 99 percent of the radioactivity resulting from the original waste emplaced in the underground facility will be retained within the set of waste packages.

3. Any releases from the waste packages that occur during the containment period should be gradual such that releases from the engineered barrier system in any 1 year during this period should not exceed 1 part in 100,000 of the total inventory of radionuclide activity present in the geologic repository system in that year.

2.2.2.2 Gradual Release of Radionuclides

The NRC regulations establish a performance objective for the gradual release rate of radionuclides from the engineered barrier system following the end of the containment period. The requirements of 40 CFR Part 191 state that the release rate control period will end at 10,000 years after closure.

The portion of 10 CFR Part 60 that sets the performance objective for control of radionuclide release rates is Section 60.113(a)(1)(ii); it states, in part, the following:

> ...the engineered barrier system shall be designed, assuming anticipated processes and events, so that (B) The release rate of any radionuclide from the engineered barrier system following the containment period shall not exceed one part in 100,000 per year of the inventory of that radionuclide calculated to be present at 1,000 years following permanent closure, or such other fraction of the inventory as may be approved or specified by the Commission; provided that this requirement does not apply to any radionuclide which is released at a rate less than 0.1 percent of the calculated total release rate limit. The calculated total release rate limit shall be taken to be one part in 100,000 per year of the inventory of radioactive waste, originally emplaced in the underground facility, that remains after 1,000 years of radioactive decay.

DOE's Office of Geologic Repositories has interpreted the boundary of the engineered barrier system as follows:

> The engineering barrier system boundary is defined by the envelope of the underground facility. For the purposes of the site characterization plan, the boundary for evaluation of releases from the engineered barrier system relative to the requirements of 10 CFR 60.113 is different than the envelope of the underground facility and is conservatively chosen to coincide with the surfaces of the excavations within the underground facility. In making this evaluation, the release rates calculated are those corresponding to the net flux of radionuclides transported into the host rock.

The waste package for the SRP interfaces with the surface of the excavated emplacement hole, thus for the SRP the external surface of the waste package corresponds to the boundary of the engineered barrier system.

2.2.3 Conceptual Waste Package Design

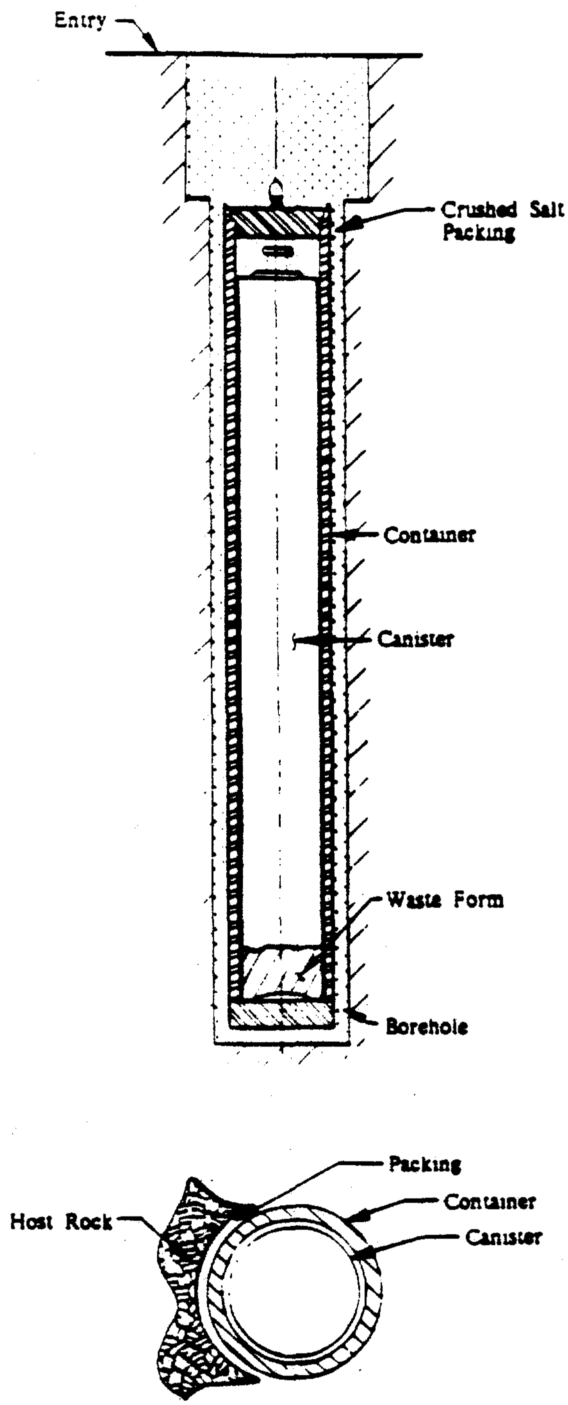
The waste package means the waste form and any containers, shielding, packing, and any other absorbent materials immediately surrounding an individual waste container (see 10 CFR 60.2). The waste package components for the SRP conceptual designs described in this section are the waste form, a thin-walled metal canister, a thick-walled low-carbon steel container, and a packing between the container and the host rock. The waste forms are described above in Section 2.2.1. Figure 2-1 shows the reference conceptual design for the waste package and identifies its components. The conceptual design studies are described by Westinghouse (1986) and by ONWI (1988a).

2.2.3.1 Canisters

The canister is the thin-walled metal receptacle which holds the waste. The major purpose of the canister is to contain the waste during the operations which occur from the time the waste is placed in the canister until the canister is inserted into the container during the packaging process at the repository. The canister's only postclosure function is to assist in transferring heat from the spent fuel rods in the consolidated spent fuel package. The dimensions of the canisters are given in Figure 2-2. During the solidification of liquid HLWs at both the Savannah River Plant and the West Valley Demonstration Plant, the canister will serve as the mold into which the molten glass waste will be cast and subsequently cooled. Each canister will be sealed by welding and stored until shipped to the repository for disposal. Because of the elevated temperatures in an air environment during the glass casting and cooling process the canister will be fabricated of Type 304L stainless steel to minimize oxidation. The dimensions of the West Valley HLW canister are equal to or less than those of the defense high-level waste (DHLW) canister.

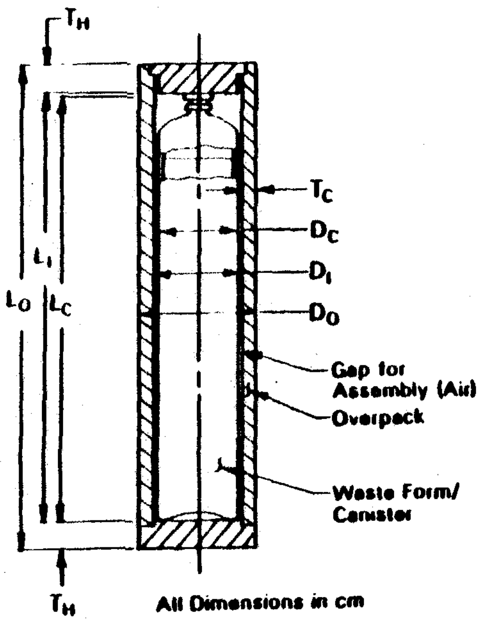
Intact Spent Fuel Canisters. The canisters for intact spent fuel are fabricated from low-carbon steel. A canister will hold up to four intact PWR fuel assemblies (or four boxes of reactor-consolidated PWR fuel) or nine intact BWR fuel assemblies (or nine boxes of reactor-consolidated BWR fuel). Each canister consists of four components: a baseplate, a cylindrical shell, a fuel cage, and a cover plate with a lifting pintle. A different fuel cage design is necessary for PWR and BWR assemblies. The cage is made of three circular plates, each having a square hole to position a 2 by 2 array of PWR assemblies or a 3 by 3 array of BWR assemblies. The 0.3-cm-thick plates are spaced longitudinally in the canister by long steel rods. The cylindrical shell is fabricated from 0.6-cm-thick steel.

Consolidated Spent Fuel Canisters. The canister for consolidated spent fuel is segmented into six congruent compartments. Each compartment will hold the rods from either two PWR or five BWR assemblies. The radial compartment dividers not only serve as walls to facilitate incremental filling of the canister, but are effective heat-conduction paths which reduce the temperature in the interior of the canisters.



Conceptual Design for the Waste Package

Figure 2-1



WASTE FORM	DHLW	SFA(4PWR)	SFA(9BWR)	CSF(12PWR)	CSF(30BWR)
Power, watts	470	2,200	1,620	6,600	5,400
Canister					
D_C	61.0	67.3	67.3	62.0	62.0
L_C	300.0	424.2	424.2	400.0	435.1
Container					
D_I	63.5	69.2	69.2	64.5	64.5
D_O	84.8	94.8	94.8	88.4	88.4
L_I	305.0	446.0	446.0	414.5	448.1
L_O	338.3	484.6	484.6	451.1	484.6
T_C	10.7	12.8	12.8	11.9	11.9
T_H	16.8	19.8	19.8	18.3	18.3
Weights, kg					
Waste Form	1,470	2,670*	2,485*	8,000*	8,280*
Canister	470	500	500	470	500
Container	7,250	13,520	13,529	11,340	12,060
Total, Pkg.	9,190	16,690	16,505	19,810	20,840

*Representative values only. All spent fuel values will vary with vendor.

Waste Package Dimensions

Figure 2-2

2.2.3.2 Containers

The container is composed of a cylindrical body with two flat heads as shown in Figure 2-1. The physical size of the waste containers is a function of the size of the waste form it will contain and the wall thickness necessary to meet the containment requirements. The waste package container has a major role in providing substantially complete containment for up to 1,000 years. The container is also expected to affect the chemical environment of the waste package after the containment period and thereby assist in controlling the rate of dissolution of radionuclides from the waste form.

During Phase 1 only a one-length container will be used to hold either intact PWR or BWR assemblies. In Phase 2 a one-length container will be provided for consolidated PWR fuel rods and a longer container will be used for PWR assemblies that cannot be disassembled and for consolidated BWR rods.

The design thickness for the cylindrical wall and the top and bottom heads of the container is the sum of the structural thickness required to sustain the repository lithostatic pressure and the thickness of the wall predicted to be consumed by general corrosion during the first 1,000 years after emplacement. Both the structural capacity and the corrosion performance are properties of the container material. The material selected for the container is a cast low-carbon steel, ASTM A216, Grade WCA. The structural design of the container is based upon rules of the ASME Boiler and Pressure Vessel Code.

Experimental data available to date indicate that general corrosion is the principal corrosion process acting on low-carbon steel during the containment period. Localized corrosion processes are not expected to be active to cause penetration of the container, and no pitting, crevice corrosion, stress corrosion cracking, or hydrogen attack have been observed in the experiments conducted under simulated repository conditions. Low-carbon steel is believed to corrode actively under the range of expected conditions and therefore, should not be subject to localized forms of attack that are associated with a breakdown in passivity.

The prediction of the container penetration by general corrosion under expected conditions is based on the extrapolation of the rate of ferrous corrosion in salt with anoxic brine, accounting for the reduction of the container surface temperature. The anoxic brine condition will result from the consumption of the residual oxygen by the oxidation of iron after the repository is sealed. The rate of general corrosion was determined from the results of experiments at 150°C with as-cast low-carbon steel in contact with crushed salt saturated with high-magnesium brine for periods up to 1 year. The prediction uses the corrosion rate occurring at 1 year extrapolated to 1,000 years and assumes that after 1 year the corrosion rate decreases only as the container temperature decreases with an Arrhenius-type behavior. Using the predicted container temperatures, corrosion depths were calculated for three waste containers. The values were added to the respective wall thicknesses necessary for structural purposes to obtain the total wall thicknesses for the three container designs. The dimensions of the containers for the three waste types are summarized in Figure 2-2.

2.2.3.3 Packing

The packing is the material that fills the space between the container outer surface and (1) the inner surface of the emplacement hole and (2) the projection of the entry surface over the emplacement hole. The packing serves as a radiation shield between the container and the emplacement entry, is a heat conduction path from the container to the host rock, and will contribute to controlling release of radionuclides from the waste package. Crushed salt is the designated packing material for the reference design concept.

During the course of the conceptual design study, the role of the packing was changed from a secondary function for heat transfer and radiation shielding to a primary function of limiting release of radionuclides. As a result, it was necessary to develop a base of information for the packing design evaluation and for material specifications for the packing. Experimental studies were planned to gather data on potential packing materials including salt. The properties of primary interest are impedance to fluid flow, filtering of colloids, resistance to diffusion, and affect on the local chemistry environment (pH, oxidation state, magnesium concentration, etc.) The design definition and material specification for the packing was to be developed in subsequent design studies.

2.2.4 Design Assessment

This section gives a summary of the results of analyses performed to assess the thermal, structural, and radiation design properties of the conceptual waste package design.

2.2.4.1 Thermal Assessment

Thermal analyses were performed for waste packages for defense HLW, intact spent fuel, and consolidated spent fuel emplaced in a single row along the center of the floor of an emplacement entry with a maximum areal heat load of nominally 10 W/m^2 (40 kW/acre) for spent fuel and 4.84 W/m^2 (19.6 kW/acre) for defense waste. (Constraints on physical placement determine the areal thermal loading for DHLW waste packages.) For the 12-PWR consolidated spent fuel (CSF) package with a heat output of 6.6 kW (the maximum heat output package), the peak fuel cladding temperature is predicted to be 370°C about 1 year after emplacement, and 85°C after 1,000 years. The peak host rock temperature is predicted to be 135°C about 3 years after emplacement. Prior to consolidation of the crushed salt packing, the peak container temperature will be about 55°C higher than the host rock temperature, but after consolidation, the container temperature will approach that of the salt. For DHLW, the temperature of the salt at the borehole-package interface peaks at about 50°C about 20 years after emplacement. The waste centerline temperature reaches a maximum of approximately 80°C 2 years after emplacement, but by the end of the containment period the temperatures have declined to about 30°C.

2.2.4.2 Structural Analysis

The thickness of the container is designed to the conservative rules and data of the ASME Boiler and Pressure Vessel Code. An additional thickness is provided to allow a reduction in the wall thickness by general corrosion while retaining the structural capacity of the container wall. While intact, the corrosion allowance increases the structural capacity of the container beyond that necessary to sustain the lithostatic pressure.

2.2.4.3 Radiation Analysis

The radiation levels at the side surfaces of the container were obtained by computer code calculations of gamma ray and neutron transport through steel from a DHLW waste form and a 12-PWR CSF waste form. The results of these calculations are given in Table 2-2. Radiation levels around a backfilled borehole for emplaced DHLW and a 12-PWR CSF package were also estimated by similar methods. The 30-BWR CSF package will have similar or slightly lower doses than the PWR case because it has a lower radiation source term. The dose rate at the floor of the emplacement entry above a 12-PWR CSF package was estimated to be less than 0.01 mrem/hr with 1.5 m of crushed salt above the top of the container.

Table 2-2. Radiation Levels at the Side Surfaces of the Waste Packages

Waste Package	Neutrons (reflected), mrem/hr	Gamma, mrem/hr	Total, mrem/hr
DHLW	$9 \pm .5 \times 10^{-2}$	$2 \pm .5 \times 10^{3}$	$2 \pm .5 \times 10^{3}$
12-PWR CSF	$5 \pm .5 \times 10^{1}$	$7 \pm 3 \times 10^{3}$	$7 \pm 3 \times 10^{3}$

2.2.5 <u>Summary Licensing Strategy</u>

The SRP has developed a strategy for showing achievement of the postclosure performance requirements of the NRC (discussed above in Section 2.2.2). The strategy is to allocate performance to the components of the waste package. Each allocation is defined as a performance measure for the component. One or more goal values are assigned to each performance measure. Meeting the goal values will be necessary to show that the requirements will be achieved. Technical evidence will be presented to show with reasonable assurance that the goal values will be met.

This section gives a summary of the strategies for showing that substantially complete containment and gradual release will be achieved. These strategies are described in Sections 8.2.2.2.4 and 8.2.2.2.5 of the SCP (DOE, 1988).

2.2.5.1 Substantially Complete Containment

The DOE has established three design objectives to achieve substantially complete containment, see Section 2.2.2.1. The SRP has developed performance measures for the waste package components which are within DOE's design objectives. Meeting the goals of these performance measures will show that the waste package can achieve substantially complete containment. The performance measures and goals are summarized as follows:

(1) The containers for at least 90 percent of all waste packages emplaced in the repository will remain intact and retain all of their radioactive waste for at least 1,000 years after closure of the repository.

(2) The release from any failed spent fuel container will be <0.02 of its total radioactive inventory during the first year after container failure, and will be $<10^{-4}$/year of the total radioactivity present in that container during any subsequent year. The release from any failed container of HLW glass will be $<10^{-4}$/year of the total radioactive contents of that container in that year.

(3) The release of radioactivity from the packing to the host rock will be $<10^{-5}$/year of the total radioactive inventory of its failed container after the packing consolidates.

Thus, in achieving substantially complete containment, it is recognized that some container failures may occur, but the waste form will limit the release of radionuclides from the failed containers to the host rock to a very small fraction of the total radioactive inventory of the repository in any year. Later, the packing will act as a redundant barrier to the release of radionuclides to the host rock.

2.2.5.2 Gradual Release

After the containment period, the NRC requires the release of radionuclides from the engineered barrier system to be $<10^{-5}$ per year of the quantity of each radionuclide calculated to be present 1,000 years after closure of the repository. As discussed in Section 2.2.2.2, the DOE has interpreted the boundary of the engineered barrier system to be the surface of the excavated regions in the repository. Since the outer surface of the waste package, i.e., the packing, contacts the excavated surface of the emplacement hole, the outer surface of the waste package coincides with the surface of the engineered barrier system.

The quantity of water (as brine) available to the waste package is expected to be limited and may be consumed by reaction with the steel container so it will not be available to dissolve or transport radionuclides from the waste package. However, without site-specific data to support this expectation, it is the strategy of the SRP to allocate performance to the waste form, the container, and the packing to meet the controlled release requirement. During the first year after container failure, the goal is that the release from spent fuel will be <0.02 of each of those elements that are present in the fuel-to-cladding gap and in the fuel grain boundaries. For both spent fuel (after the first year) and HLW glass, the goal is to release $<10^{-4}$ per year of each radionuclide from the waste form by a combination of slow matrix dissolution, low radionuclide solubility, and/or limited concentration in the solution. The goal for the transport of radionuclides through the packing is that it be by diffusion with $<10^{-5}$ per year of each radionuclide released to the host rock. Even after failure, the goal for the steel container is to continue to corrode and consume oxidants, maintaining a reducing condition in the waste package to limit dissolution of the fuel matrix and dissolution of many radionuclides.

The SRP developed an extensive plan of investigations to collect the information and data to show that the above strategies would lead to achieving substantially complete containment and controlled release of radionuclides. These investigations would need to be conducted by in situ and laboratory tests and studies some of which would to continue through the performance confirmation period.

2.2.6 Alternative Designs

In the event that the results from research showed that the performance goals for the waste package could not be assured with the reference concept, alternative concepts were considered. These alternative concepts included changing materials and changing the design. These are briefly described below.

Consideration was given to changing the materials for the container and/or the packing. For example, corrosion-resistant alloys were planned to be investigated to allow use of a corrosion-resistant container rather than the present corrosion-allowance low-carbon steel container. Also, nonmetallic materials were considered which could be redundant to the metallic container and which could be expected to degrade by mechanisms different than those associated with metals. The use of a corrosion-resistant container material might include a design change by the use of a thin-walled corrosion-resistant material over a thick-walled steel reinforcement. (The steel reinforcement provides the required structural capacity to support the thin-walled container under the lithostatic pressure.)

Another alternative design concept which was considered was to place a low-carbon steel container outside of the above described thin-walled corrosion resistant container. The outer steel container would have a wall thickness sufficient to avoid penetration by general corrosion for 1,000 years (10 to 15 cm thick). The steel and the corrosion resistant material would be electrically coupled. If the steel container were penetrated before 1,000 years, the remaining steel would provide cathodic protection for the corrosion-resistant container. This would provide two redundant containers, either of which would be designed to last 1,000 years.

The life of the container could be increased by reducing the heat output of each package (by reducing the amount of spent fuel per container). The reduction in the maximum heat output of the waste package could result in a significant reduction in the maximum service temperature of the waste package components at the expense of increasing the number of waste packages for the same total quantity of spent fuel for disposal. Since the rate of corrosion of the steel container is reduced as the temperature is reduced, the life of the container could be extended by reducing its service temperature.

Other packing materials were planned to be investigated which could improve control of fluid flow to and from the container and waste form, lower the transport of radionuclides to the host rock, and control the local brine chemistry to reduce the rate of container corrosion or the solubility of radionuclides. Also, the use of a preconsolidated packing material could allow the packing to function earlier in the life of the waste package.

2.2.7 Pre-ACD Studies

The design phase following conceptual design is the advanced conceptual design (ACD). A contractor was selected for ACD and the activity was planned to begin in FY 1990. Prior to starting ACD, three preliminary studies were initiated to provide a basis for ACD studies. Two of these studies, a waste package components fabrication study and a remote closure weld study, were completed and their results are summarized below. The third study on postclosure inspection methods was not completed and will not be discussed further.

2.2.7.1 Component Fabrication Studies

The canister and the container were both considered in the pre-ACD fabrication studies. Since the canister has no postclosure performance allocation, its conceptual design is tentative and its fabrication will not be considered here. The container received the major attention during conceptual design and it has a major role in the postclosure performance of the waste package so its fabrication is worthy of attention.

Five sizes of containers were used for the fabrication study. Sizes range from 84.8 cm OD by 338 cm long to 94.8 cm OD by 484 cm long, with wall thicknesses ranging from 10.7 cm to 12.8 cm. Head thicknesses range from 16.8 cm to 19.8 cm.

The bottom of the container may be a separate part welded to the cylindrical shell or may be integral with the body. The upper head will feature a handling device, such as a male pintle. The handling feature will be used for positioning the head for welding and for handling the assembled loaded container. The external surface finishes of the container will be as smooth as practicable to allow easy decontamination prior to emplacement and to minimize the number of potential crevice corrosion sites. The handling feature (male pintle) will also be designed to facilitate remote decontamination. Material for all container components will be a low-carbon steel with mechanical and chemical properties equivalent to ASTM A216 Grade WCA.

Heads for both ends of the container may be machined from static castings, plate, open-die forgings, or closed-die forgings. Static castings or closed-die forgings made to near-net shape followed by machining will probably yield the lowest cost products; forging will yield the best metallurgical quality. A pintle or handling feature can be cast or forged as an integral part of the head. Detailed design of the handling feature will be performed as part of future waste package activities.

The following fabrication-process candidates have been considered for the container shells:

- Extruding
- Forging
- Centrifugal casting
- Static casting
- Laminating (spirally wrapped thin layers)
- Forming and welding.

The choices of fabrication method for the container shell are all capable of producing the required geometry. Centrifugal casting, static casting, and extrusion are the preferred methods as determined by relative complexity, with a slight bias toward casting based on cost considerations. Cost differences between casting and extrusion are not as great as originally expected and future detailed cost estimates may alter this recommendation.

Static casting the container shell and bottom head in one piece is a viable alternative that should be considered. This option does entail increased machining complexity but reduces shop welding and inspection. Material composition and microstructural differences encountered with a two-piece shell and bottom head design are avoided, reducing the potential for corrosion problems.

These recommendations were made without consideration of potential surface defects that may be associated with a candidate forming process. Acceptance criteria for surface defects (cracks, flaws, etc.) have not been established.

The selected method for the container-head surface finish is by machining and abrasive grinding with, possibly, some local buffing around the pintle. The recommended container-shell surface finishing method is machining for the inside and outside surfaces. After welding, the final surface finish will be obtained by mechanized belt grinding the container outside surface. If necessary to achieve the specified finish, a final buffing or glass-bead blasting operation may be performed at the fabricator's plant. Decisions related to contamination control and decontamination methods may alter this recommendation.

2.2.7.2 Remote Closure Welds

The closure process is required to provide a high-integrity seal of the containers. The closure must be performed under hot-cell conditions by equipment operated by remote control. Additional requirements will be imposed by the need for a highly reliable process that can be qualified on the basis of monitored process variables and remote equipment maintenance. The process may be a conventional joining process or may require developments of the state of the art.

The following processes are under consideration for the container closure welds:

- Gas metal arc weld-narrow gap (GMAW-NG) twist wire
- Gas tungsten arc weld-narrow gap (GTAW-NG)
- Electron beam weld (EBW)
- Electroslag weld (ESW)
- Submerged arc weld (SAW)
- Plasma arc weld (PAW)
- Inertia weld.

The recommended weld method selected for the container is the EBW process. Selection of the EBW process was based on its ability to produce full penetration welds in a single pass at a high production rate without the addition of filler material. Computer controlled equipment, adaptable to a remote application, further contributes to the selection of EBW as the primary container weld method.

The GTAW-NG process is recommended as a backup container weld method. This recommendation is qualified, assuming that a partial penetration weld, limited to about a 2-in-weld depth, will be acceptable. If a full penetration weld is required, the GMAW-NG twist wire process is a recommended alternate method. The GMAW process produces a slag that requires cleaning during welding. Welding heads must also be changed during the course of completing the weld. Both the GTAW and GMAW processes are considerably slower than the EBW process.

Even though the inertia weld process was rated higher than the GMAW process, it is not recommended for the container weld. Serious questions regarding availability of suitable equipment to produce a weld on a component the size of the container (machinery must be capable of producing a thrust of about 12×10^6 lb for the container inertia weld) remove this process from further consideration.

Only inertia and EBW processes were capable of satisfying the 4-hr-cycle time requirement. The other candidate weld processes, all requiring the addition of filler metal, resulted in estimated cycle times about twice as long as the proposed 4-hr value. The long cycle times are acceptable but twice as many weld stations would be required in the repository.

2.3 WASTE PACKAGE MATERIALS RESEARCH

The primary method for the disposal of nuclear wastes being considered by the DOE is placement in stable geologic formations deep underground. The mined geologic disposal system will consist of three subsystems: (1) the natural system associated with the site, (2) the repository, and (3) the waste package. Together, these subsystems provide a system of multiple, independent, natural, and man-made barriers to the release of radionuclides from nuclear waste. This section deals with the waste package, more specifically with the materials that make up the waste package, and summarizes research results that can be used by the waste package designers.

To promote technical conservatism, the program was to develop and test a long-lived, high-integrity waste package. This package was to be designed to provide "reasonable assurance" that the two pertinent performance guidelines of the NRC would be attained (see Section 2.2.1.3).

The overall objectives of the materials test program were to evaluate candidate waste package materials and demonstrate that they could meet the functional waste package requirements, as described by the DOE (1982a). The program's test results would help provide a basis for designing the package and for assessing its performance for licensing. The program was also expected to provide an experimental basis for developing models of degradation modes for specific package components. These models would be used in turn to build a complete package degradation model for use in design and performance assessment of the package. Additional experiments from this program could also help to validate such models.

The initial conceptual designs for waste packages in a salt repository have been completed (Westinghouse, 1983). The package designs for spent fuel, CHLW, and DHLW are basically the same but with varying dimensions. The designs consist of a waste form and production canister and a container. The cylindrical container is fabricated from carbon steel of sufficient thickness to allow corrosion over 1,000 years while still retaining sufficient thickness to resist cracking from external pressure loads from lithostatic pressure. Alternative designs utilize a thin overpack of TiCode-12 surrounding the carbon steel reinforcement. Current conceptual designs for salt do not include a tailored emplacement hole packing material but previous studies and recommendations for future studies on such material will also be discussed in this section.

2.3.1 Waste Form Materials

Numerous waste forms have been under development and evaluation for some time, and a large amount of materials data on these forms can be found in the literature. The waste forms include borosilicate glass, glass ceramics, silicate ceramics, high alumina ceramics, titanate ceramics, high silica glasses, cements, coated or composite ceramics and glasses, metal matrix forms, phosphate-based glass and ceramics, and spent fuel. However, the largest body of literature exists for borosilicate waste glasses and spent fuel.

Candidate HLW forms have been under investigation for a number of years and the characteristics of a variety of candidate materials over a range of test conditions have been reviewed (Mendel et al., 1981; and McVay et al., 1981). In this section we will discuss only results pertaining primarily to release characteristics in salt brines. Much early work was done in deionized (D.I.) water which provides a bases for understanding the influence of brine species on radionuclide release. Site-specific brine compositions were not available until very recently and most brine studies utilize the Brine B composition. Several sites were undergoing more detailed investigation of their suitability as a nuclear waste repository. One of these sites is in the Permian Basin in west Texas. Drill cores from there have been evaluated at Pacific Northwest Laboratory and a reference composition obtained for saturated brines made from these Permian cores. This brine composition closely resembles WIPP Brine B. Therefore, significant differences are not expected in leaching characteristics between the ONWI composite and Brine B. Also, at Savannah River Laboratory, defense waste glass was leached in the presence of Avery Island and Carlsbad salt, and no significant difference in leachability was found (Wicks et. al., 1982).

The waste forms under consideration are spent fuel and a HLW form of glass containing waste from reprocessing. Although a number of potential HLW waste form materials have been considered, the selected DHLW form is a borosilicate glass. Although a CHLW form has not yet been selected, it is likely to be a borosilicate glass also, such as PNL 76-68. Therefore only the characteristics of spent fuel and borosilicate glass will be considered in this report.

2.3.1.1 Borosilicate Glass Dissolution

The release of radionuclides from nuclear waste glasses is ultimately controlled by the rate of glass matrix hydrolysis or dissolution. The concentration of dissolved silicon in the contacting solution has been identified as the most important factor controlling the dissolution rate (Pederson et al., 1983; Chick and Pederson, 1984; Grambow, 1985). At early times of exposure to solution, glasses dissolve irreversibly at a maximum rate that depends upon temperature, glass composition, and solution composition. As the concentration of dissolved silicon in solution increases, the initial reaction rate decreases until a final reaction rate is established when the solution is saturated with respect to a silica-bearing mineral. Glass reactivity is further reduced in salt brines, compared to low-ionic-strength ground waters, perhaps due to the reduced solubility of silica, changes in dissolved silica activity coefficients, and/or the activity of water.

The final rate of glass dissolution at silica saturation is thought to be fixed by the net rate of silicic acid consumption. Silicic acid may be consumed by the precipitation of alteration phases. In dilute magnesium chloride solutions, for example, the precipitation rate of sepiolite [$Mg_2Si_3O_6(OH)_4$] determines the rate of glass reaction. The precipitating phases controlling glass dissolution in more concentrated and complex brine solutions are yet to be identified. The following sections briefly discuss the dependence of glass dissolution on several important factors including temperature, pH, redox potential, etc.

<u>Temperature</u>. The rate of glass dissolution in an aqueous medium is highly sensitive to temperature. The dissolution rate varies in an Arrhenius fashion over a temperature range of 25 to 300°C; the rate depends on the glass and solution composition. Activation energies range between 10 and 35 kcal/mol. Although high-temperature testing accelerates reaction rates, it is not clear that the reaction mechanisms are identical at more repository-relevant temperatures. This caveat is particularly applicable to salt brines because hydrolysis reactions, such as the precipitation of $Mg(OH)_2$, may significantly affect the solution pH.

<u>pH</u>. Solution pH is an important variable that affects the dissolution rate of glasses. The solution pH attained is a system-dependent variable, with glass composition, radiation field, and waste package components all affecting the pH to some extent. In low-magnesium salt brine, the solution pH increases with reaction progress from the initial neutral pH and exhibits 3 to 4 pH unit transient excursions in the presence of iron. In high-magnesium brine, the

solution pH remains relatively stable (approximately 7) regardless of the presence of iron, possibly because of the hydrolysis of Mg species. Consequently, the solution pH that may be attained in a salt brine depends not only on glass/brine interactions but also on the brine composition itself.

Redox Potential. At present, little is known about how the solution redox potential may or may not affect glass dissolution rates. If the reactions that consume and produce silica in solution are not redox sensitive, then the glass dissolution rate may be unaffected by redox conditions. However, redox potential has been shown to affect the dissolution rate of glasses in the presence of iron. Silica readily reacts with or sorbs onto ferrous and ferric iron corrosion products, thereby increasing the rate of glass dissolution.

Radiation. Radiation affects glass degradation processes through radiation damage to the glass itself and radiolysis of the solution. However, radiation-damaged glasses have not exhibited increased susceptibility to aqueous attack. On the other hand, gamma radiolysis, in the presence of an N_2 and O_2 gas phase, produces nitric acid which significantly increases the dissolution rate of a glass exposed to the resulting acidic solution. Alpha radiolysis has a small effect (factor of 3 increase) on glass dissolution rates at alpha dose rates expected for a commercial waste glass.

Brine Composition. The effects of variations in brine composition on the dissolution of glasses have been considered. A significant increase was found in the dissolution rate of a borosilicate glass in high-temperature tests (200°C) upon increasing the Mg content of the contacting brine. The observed effect was primarily due to the increased acidity of the brine with increasing Mg concentration. However, low temperature studies (90°C) show no measurable effect of brine composition (Palo Duro Basin Brine No. 1 (PBB1) versus Palo Duro Basin Brine No. 3 (PBB3) on the glass dissolution rate.

Container Materials. The presence of iron-bearing materials in dissolution tests is known to increase glass dissolution rates under oxic condition. Iron or iron corrosion products are thought to consume dissolved silica, thereby increasing the driving force for glass dissolution.

Under appropriate hydrodynamic, pH, and redox conditions, the rate of radionuclide release from the waste package is directly proportional to the solubility of the radionuclide. Consequently, the solubilities of radionuclides released from borosilicate waste glass are extremely important to repository performance assessment.

Colloids present a major uncertainty in solubility-limited mass transport analyses (Chambre et al., 1982) because solution concentrations can far exceed the solubility limits of a species if the species is also present in colloidal form. The formation of colloids in aqueous systems is a much studied phenomenon, but little work has been performed in high-ionic-strength solutions such as salt brines. Large increases in the centrifugeable fraction of Am, Pu, and Np upon increasing the ionic strength of the sodium perchlorate ($NaClO_4$) solution between 0.01 and 1 M have been reported. The formation of Am and Pu colloids in 5 M sodium chloride (NaCl) from photoacoustic spectral measurements and ultrafiltration techniques revealed that microcolloidal (=10-A diameter) polymeric species were present. Pu(IV) in seawater samples was primarily associated with particles less than 0.22 um in diameter. Twenty-five percent of the Am released in brine in glass dissolutioning experiments was retained on a 0.1-um filter. These results clearly indicate that actinides can exist as particulate species in high-ionic-strength solutions.

2.3.1.2 Spent Fuel Dissolution

The dissolution of radionuclides from spent fuel is widely accepted to be divided into three components originating in (1) the fuel-cladding gap, (2) the grain boundaries, and (3) the uranium dioxide (UO_2) matrix. Important aspects of these three components of dissolution are further described in the following paragraphs.

The main radionuclides found within the gap and grain boundaries are the fission products Cs and I. In addition, Tc seems to be concentrated somewhat in the gap and grain boundary regions (Reimus and Simonson, 1987; Van Luik et al., 1987). Because of the high solubility of these elements and with the greater accessibility of these regions to the solution, the dissolution rate from the gap and grain boundaries is much greater than from the matrix. Thus, the dissolution rate falls off with time as the inventories in first the gap region and then the grain boundaries are depleted; it is expected to eventually reach a steady-state value representative of dissolution of the UO_2 matrix itself. The time required to deplete the gap inventory is on the order of a few days. A conservative approach, therefore, is simply to assume that the entire inventory of the gap is dissolved instantly upon failure of the container. Thus it is important to have a measure of the radionuclide inventories located within the gap region.

It is expected, but not yet demonstrated, that the actinides and fission products, even Cs and I, located within the UO_2 matrix will dissolve more or less congruently with uranium, because it will be necessary to dissolve the UO_2 matrix to expose these other elements to the solution where they also have an opportunity to dissolve.

The following sections briefly describe what is known about the effects of potentially important parameters on the release of radionuclides from spent fuel grain boundaries and matrix and/or the dissolution of UO_2 in aqueous solutions.

Temperature. Studies of temperature effect on dissolution rates of spent fuel or UO_2 in deionized water and low-ionic-strength ground waters have yielded seemingly conflicting results. Activation energies ranging from 15 to 59 kJ/mol have been determined.

Little or no effect of temperature in the range 25 to 75°C on the release of U, Pu, Tc, and Cs (the only elements studied) from spent fuel into PBB1 salt brine has been found. When temperatures were increased to 90°C in subsequent studies, Cs release was found to be about 3 times greater than at 25 to 30°C, and the release of U and Tc was also found to be higher, but by a somewhat smaller factor. The release of Pu was unaffected.

pH. The effect of pH on radionuclide release from spent fuel has not been studied in detail because most spent fuel studies have used the solution of choice without any systematic variation in the pH. The pH of PBB1 brine does not change during spent fuel dissolution studies or as a result of gamma irradiation. PBB3 brine, on the other hand, does undergo a reduction in pH during spent fuel dissolution tests, possibly as a result of the associated radiation field. However, pH has not been a controlled variable in any of the brine tests.

Oxidation Potential. The solubility of UO_2 in deionized water is extremely low. Generally, therefore, UO_2 dissolution is preceded by a sequential oxidation of UO_2 to UO_{2+} to U_4O_9 to U_3O_7, etc., with the eventual formation of U(VI) in solution. Since the oxidation of UO_2 by water is thermodynamically unfavorable, its oxidation requires the presence of dissolved oxygen or other oxidant. Spent fuel and UO_2 dissolution rates increased with dissolved oxygen concentration according to a first-order rate dependence in both deionized water and low-ionic-strength ground water and the dissolution rates of Sr, Pu, and U from spent fuel decreased with decreasing oxygen concentration, but not to the extent that would be expected if a first-order dependency of rates on oxygen concentration were obeyed.

Radiation. Radiation, both gamma and alpha, is expected to increase the oxidation potential of brines. Water is decomposed by radiation to produce a number of primary radiolytic products consisting of both reducing and oxidizing species, i.e., H_2, H, H^+, e^-aq, OH, and H_2O_2. Although the reducing and oxidizing species are generated in equal stoichiometric quantities, hydrogen (the primary reducing species) is relatively inert (in the absence of catalysts) at the temperatures expected in the repository, whereas the oxidizing species (e.g., H_2O_2) are strong oxidizing agents. Thus, the net effect of radiation is expected to be an increase in the oxidation potential of brine. These expectations are supported by some direct Eh measurements made on alpha-irradiated brines. Eh values of +1.2 volts on brines that had received total alpha doses of 1 to 4 Mrad from dissolved curium-244 were measured.

Brine Composition. Release rates of various radionuclides from spent fuel have been studied in a variety of solutions. Commonly used solutions have included deionized water, solutions that simulate various ground waters (including salt brines), and carbonate/bicarbonate solutions. In general, solution composition has been observed to have a greater effect on the release rates of radionuclides that are incorporated in the UO_2 matrix than on the release rates of radionuclides that are present at the exposed surface of the fuel (e.g., Cs and I). This suggests that the main effect of solution composition is on the degradation rate of the UO_2 matrix.

The effect of different saturated salt brines on the dissolution of radionuclides from spent fuel showed that the release of U, Tc, and Cs from spent fuel into PBB1 and PBB3 brines was similar. However, the release of Pu in PBB3 brine was somewhat greater than in PBB1 brine.

Waste Package Components. Ductile cast iron coupons were included in some of the spent fuel and unirradiated UO_2 dissolution tests conducted in salt brine and deionized water. The presence of the iron coupons in the UO_2 and the spent fuel tests in brine had the effect of reducing solution concentrations of U, Pu, and Tc but did not affect the total amounts of these elements that were released (most of the released material was in particulate form or deposited on the container walls or iron coupons). Iron had no effect on the cesium. In the UO_2 tests that were conducted in deionized water, the iron coupons reduced both the solution concentrations of U and the total amount of U that was released.

2.3.2 Container Materials

Major concerns with regard to the containment of radionuclides in a repository in salt are corrosion of the container alloy by brine, and collapse due to the in situ lithostatic pressure. A broad spectrum of candidate materials has been evaluated by screening programs, both in the USA and elsewhere, relative to performing one or both of the above functions in salt. Among the classes of materials evaluated were the following:

- Metals and alloys
- Ceramics
- Carbons and graphites
- Cements
- Organic compounds and polymers.

A summary of screening work in the USA may be found in Appendix A of DOE (1982b). From this work, and as an outcome of a number of conceptual waste package design programs, a broad consensus developed in favor of the use of metals and alloys for the primary structural and corrosion barriers (container and structural reinforcement component). While many of the other classes of materials possess at least some attributes that rank well against the corresponding ones for metals, none of them are without at least one serious drawback. A dominating concern with most of the nonmetals is the expected difficulties in their fabrication into a hermetically sealed configuration; the ability of most metals and alloys to be welded was an important factor in their selection over the other material classes. The brittle nature of many nonmetallic materials and concern about irradiation effects on organic materials were also important factors in their elimination.

A summary of some of the corrosion screening studies on metals and alloys has been provided by Merz (1982). A summary of these results is provided in Table 2-3. The original data were obtained from PNL (Westerman, 1980) and Sandia National Laboratories (Braithwaite and Molecke, 1980). Both sets of data were obtained by testing in deaerated WIPP Brine A at 250°C. However, the Sandia National Laboratories (SNL) test duration was 28 days, while the PNL data were generally obtained over 58 days. Also, in the PNL tests, the specimens were all placed in the same autoclave, and most likely the rapid iron and copper corrosion rates had a suppressing effect on the more passive materials such as Hastelloy C-276. The PNL results indicate a significant contrast in the corrosion rates of cast iron, copper, copper-nickel, and nickel to those of 304 SS, nickel-base superalloys, or TiCode-12 - as much as 1,000-fold. The trend in the SNL data is similar except, perhaps, for copper and the copper-nickel alloy. However, further testing of 90 Cu-10 Ni revealed the formation of a poorly adherent, heavy oxide scale (Molecke et al., 1981). A similar behavior with copper would be expected, as was indicated by the longer term PNL results. The corrosion rates noted for 1018 steel (SNL) were quite similar to that of cast iron (PNL).

Uniform corrosion data alone could not be used to preselect a metal for the container application. Other selection criteria were formulated to assist in further shortening the list. These criteria included the following:

1. Avoid creating an underground deposit of materials which may, someday, be attractive for salvage reasons.

2. Avoid depleting supplies of critically strategic materials.

3. Limit prolonged and expensive research and development programs which may delay implementation.

4. Take advantage of historical performance experience with those technologies which might be incorporated into repository performance models.

5. Specify production and fabrication processes which have proven to be amenable to inspection and control from a quality assurance (QA) standpoint.

6. Consider factors of cost in tradeoffs between design approaches meeting the requirements for engineered waste package design. Although cost criteria are difficult to specify, design efforts had to recognize that because of the large amount of nuclear waste to be isolated, both existing and projected, cost per unit of mass is a significant factor.

Table 2-3 Corrosion Rates in Screening Test in WIPP Brine A[a] at 250°C

Material	Corrosion Rate, mm/yr	
	PNL	SNL
304L Stainless Steel	0.0003	0.018
410 Stainless Steel	0.023	- - -
Lead	- - -	0.5
Copper	0.414	0.07
90 Cu-10 Ni	- - -	0.14
70 Cu-30 Ni	0.203	- - -
Nickel 200	0.034	- - -
Monel 400	- - -	0.03
Ductile Cast Iron	1.39-1.95	- - -
1018 Steel	- - -	1.7
Inconel 600	0.0009	0.009
Hastelloy C-276	0.00042	0.007
TiCode-12	0.00019	0.003
Zircalloy-2	0.0022	0.001

Sources: Merz (1982); Westerman (1980); and Braithwaite and Molecke (1980).

a. WIPP Brine A is a high-magnesium brine.

For a salt repository, where a combination of corrosion resistance and structural integrity at both high temperatures and high lithostatic loads is required in the overall waste package design, the material candidates tended to fall into two categories: (1) relatively inexpensive and readily fabricable, but with moderate to poor corrosion reistance, and (2) relatively expensive and generally more difficult to fabricate, but with moderate to excellent corrosion resistance. This led to consideration of two design approaches for conceptual waste package designs for a repository in salt.

One approach was to provide the necessary structural properties by use of a category (1) structural member overlaid with a relatively thin skin of a category (2) highly corrosion-resistant material. The second was to provide both the structure and corrosion barrier using a relatively thick member of a category (1) material. The low cost and fabricability compensates for the increased thickness of barrier material required.

The titanium alloy TiCode-12 was selected as the prime, highly corrosion-resistant material based on performance and cost analyses (Braithwaite and Molecke, 1980). While the 304L stainless steel appears quite attractive with a very low corrosion rate, it is well known that this material will be subjected to stress corrosion cracking when oxygen is added to the chloride environment to simulate radiolysis effects or by air access to the repository. The nickel-base superalloys, e.g., Hastelloy C-276, all contain significant quantities of strategic elements and they are of high cost. However, they were adopted as a backup for TiCode-12 as the category (2) material.

The selected candidate for the category (1) type material is low-carbon steel (also known as mild steel). This material was selected over its competitors based on cost and historical experience including weldability, structural usage, and other available technologies. In addition, low-carbon steels have the advantage of exhibiting a predictable uniform corrosion rate and a very low susceptibility to stress corrosion cracking and other localized, nonpredictable forms of environmental attack.

Based on the criteria presented above, a plain carbon steel was selected for the container material to be used in a salt repository. Moreover, since the bulk of the container will likely be produced by centrifugal casting, ASTM Casting Specification A216-77, Grade WCA, was selected as the material for testing. This grade contains 0.25 percent carbon maximum and has a minimum yield strength of 30,000 psi, a minimum ultimate strength of 60,000 psi, an elongation of 24 percent minimum, and a minimum reduction in area of 35 percent.

2.3.2.1 Ferrous Alloys

Evaluation of the corrosion performance of mild steel under salt repository conditions has consisted of (1) anticipating its failure modes by means of an in-depth survey of the literature (Westerman, Pitman, and Rhoads, 1986), (2) assessing the available degradation data, and (3) performing laboratory-scale experimental tests to determine its corrosion behavior in test environments similar to those expected in a salt repository. The degradation modes receiving the primary attention in the SRP are general or uniform corrosion and certain forms of non-uniform corrosion, and stress corrosion cracking. These studies are discussed in the following sections.

General Corrosion. The general corrosion behavior of mild steels has been characterized by electrochemical studies and by autoclave-type corrosion tests. Electrochemical techniques have been used primarily to study corrosion mechanisms and to determine corrosion rates. The autoclave-type tests were used to simulate a wide range of repository conditions through adjustments of the controlled variable associated with planned test matrices (Westerman et al., 1987). In autoclave-type tests the response of the material (corrosion penetration, uniformity of corrosion, nature of corrosion product, etc.) to the test environment is assessed after the test exposure is completed.

The general corrosion investigation of A216 steel performed by the SRP was directed primarily toward determining the effect of the following environmental variables on the corrosion behavior of the material to provide a data base for a predictive model:

- Mg^{++} concentration
- Temperature
- Time
- Hydrogen overpressure
- Irradiation intensity.

The major efforts were directed to the first three parameters, and these will be discussed in further detail. The average corrosion rate was found to decrease approximately linearly with increasing the overpressure. The rate observed at an H_2 overpressure of 21 MPa (3,000 psig) was only ~20 percent of that observed when the H_2 overpressure was near zero. The retention of high-pressure hydrogen within a repository could therefore inhibit the corrosion of a mild steel waste package container. Irradiation intensity at levels expected at the container surface with the current waste package designs were found to have no effect on the corrosion rate.

Effect of Mg^{++} Concentration on Corrosion Rate. Tests of mild steel in anoxic low-Mg "dissolution" brines (such as PBB2 brine, having ~100 ppm Mg^{++}) at 150°C showed nominal average corrosion rates of less than 25 um (1.0 mil) per year. Subsequent testing of the material in low-Mg two-phase (excess-salt) environments corroborated the initial brine results. For example, when A216 (Lot 1)* steel specimens were tested in an environment of PBB1 salt with PBB1 brine (added to bring the water concentration to 20 wt %) for a period of 12 months at 150°C, the average corrosion rate observed at the end of the 12-month period was ~3 um/year. In all low-Mg^{++} tests the corrosion product was magnetite, Fe_3O_4.

The general corrosion rates of mild steel in chloride brine or salt/brine environments have been found to be markedly enhanced by as much as two orders of magnitude, by the presence of Mg^{++} (Molecke et al., 1982; Westerman et al., 1986; Westerman et al., 1987). High-Mg^{++} test environments have been emphasized because such tests provide the most severe (conservative) tests of

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* Lot 1 contains about 0.25 percent carbon and Lot 2 contains about 0.15 percent carbon.

the material with respect to general corrosion, and because it is not possible at the present time to exclude the possibility that high-Mg brines might exist in the waste repository.

The specific dependence of the general corrosion rate of A216 steel on Mg^{++} concentration in Na-Mg chloride brines has been determined. Specimens of A216 steel, in the as-cast and normalized conditions, were exposed to PBB3-like brines containing varying quantities of Mg^{++} (i.e., 1,000, 5,000, and 10,000 ppm), in addition to the test exposure using high-Mg^{++} PBB3 brine, containing 48,000 ppm Mg^{++}.

The average corrosion rates of A216 Lot 1 specimens over 1- and 6-month test durations in variable-Mg^{++} brines showed an increase of corrosion rate with Mg^{++} concentration that is essentially linear under these test conditions. A^{+} the high-Mg^{++} concentrations associated with PBB3, the normalized material corrodes at a higher rate than the as-cast material, a common finding, and the corrosion product (identified post-test by X-ray diffraction analysis) is always complex iron-magnesium hydroxide with an amakinite structure. At Mg^{++} concentrations of 0 to 10,000 ppm, the corrosion product retains a magnetite (Fe_3O_4) structure. There was some diminution of corrosion rate with time over the 6-month duration of the test. The data showed the importance of Mg^{++} concentration in the corrosion of steel in sodium chloride brines containing Mg^{++}.

<u>Effect of Temperature on Corrosion Rate</u>. The effect of temperature on the corrosion rate of A216 steel specimens was determined in an all-brine (PBB3) environment and a high-Mg excess-salt test environment.

The effect of the specific material conditions, i.e., as-cast and normalized, on the corrosion rate was found to be minimal at 90°C; however, at 150 and 200°C an effect became apparent, with the normalized material corroding at the higher rate. The corrosion rate of the as-cast Lot 1 material was found to increase by a factor of approximately 100 between temperatures of 90 and 200°C, and by a factor of 5 between temperatures of 150 and 200°C. Assuming an Arrhenius relationship between corrosion rate and test temperature, the activation energy corresponding to the corrosion rate enhancement with temperature is approximately 15 kcal/mol. The constant activation energy over the temperature range investigated, i.e., between 90 and 150°C, and between 150 and 200°C, is consistent with one rate-controlling mechanism.

If an Arrhenius-type relationship is presumed to apply, the 3- and 6-month data in the excess-salt test environment suggest the possibility of a corrosion mechanism change with temperature, since the apparent activation energy associated with the rate increase between 90 and 150° lies, for all of the data, between 10 and 15 kcal/mol, whereas between 150 and 200°C the activation energy for all of the data is much lower, lying between 4 and 6 kcal/mol. The high activation energy is consistent with a charge transfer process, such as that associated with H_2O reduction at cathodic sites. The low activation energy value is consistent with diffusion-transport processes.

<u>Effect of Time on Corrosion Rate</u>. The results of tests on 216 (Lot 1) steel specimens in PBB1 salt with PBB3 brine at 150° for times up to 1 year are given in Table 2-4.

The corrosion rate can be seen to diminish with time, with the average rate becoming as low as 0.25 mm/year (10 mil/year) after 12 months. Specimens removed from the test can were surrounded by thick, gray, clay-like layers of corrosion product, held in place by the surrounding excess-salt environment. An X-ray diffraction analysis identified the corrosion product as amakinite. Thus, the corrosion rates are high relative to low-Mg brines, but tend to decrease with time, apparently either from the protection (reactant diffusion control) offered by the thickening corrosion product layer, which is held in place at the specimen surface by the surrounding salt phase, or from diffusion rate-control imposed by the salt matrix surrounding the specimens.

Table 2-4. Mean Corrosion Rates of A216 (Lot 1) Steel Specimens in PBB1 Salt With PBB3 Brine, 20 wt % H_2O at 150°C

Material treatment	Mean Corrosion Rate, mm/yr (mil/yr) 1-mo test	3-mo test	9-mo test	12-mo test
As-cast	0.62 (24)	0.67 (26)	0.42 (17)	0.25 (10)
Normalized	0.77 (30)	-	-	0.61 (24)
Homogenized	0.92 (36)	-	-	0.60 (23)

To further define the nature of the corrosion products formed in high-Mg excess-salt test environments, a study was undertaken to determine whether there were marked differences in composition as a function of depth in the corrosion product layers formed under these conditions. Differences could signify diffusional concentration gradients, presence of different reaction product compounds, ranges of stoichiometry of single compounds, or all three. The corrosion-product layers formed on the specimens in the 12-month test were serially sectioned with a knife blade into four sublayers per surface. Chemical analysis of the layers revealed that the corrosion product layer samples were high in Fe and Mg, compared with the composition of the environment sample at a distance from the specimen, and correspondingly low in Na and Cl. The Mg concentration was essentially constant throughout the corrosion product layer, suggesting the generation of a single principal reaction product of essentially invariant composition throughout the entire course of the reaction. If the reaction product layer is assumed to consist of amakinite $Fe_xMg_y(OH)_2$ only, as indicated by X-ray diffraction analysis, the corresponding chemical formula for the corrosion product formed is $Fe_{0.64}Mg_{0.36}(OH)_2$, or approximately $Fe_{2/3}Mg_{1/3}(OH)_2$.

The 12-month excess-salt test contained normalized and homogenized (930°C, 24 h, air cool) specimens of A216 steel in addition to the as-cast material, to determine what effect such heat treatments might have on the steel's corrosion resistance in high-Mg environments. This evaluation was considered important because an austenitizing treatment, whether purposefully applied for mechanical property optimization or as an unavoidable process occurring in the heat-affected zone of any closure-welding operation, could be a factor in the final waste package design considerations. It was found that the as-cast material corroded at the lowest average rate - ~0.25 mm/year (~10 mil/year) after 12 months - as noted in Table 2-4, whereas the heat-treated materials corroded more rapidly, exhibiting penetration rates as high as 0.75 mm/year (30 mil/year). It is not clear whether the rate reduction of the as-cast material resulted from diffusion control, either by the corrosion product layer or by the salt-brine matrix, or whether the rate reduction was caused by a diffusion control coupled with anode/cathode activation control (charge transfer processes). Although a buildup of corrosion product would eventually stifle the corrosion of the steel, the rate of diminution seen over the course of the 12-month test was not nearly as significant for the austenitized material as it was for the as-cast material, suggesting a strong microstructural influence on corrosion kinetics. Only tests of significantly longer duration would reveal whether a satisfactorily low corrosion rate would eventually be obtained for all heat treatments.

Nonuniform Corrosion. Artificially pitted (drilled) specimens and crevice specimens consisting of bolted-together plates were exposed to a two-phase salt-brine environment composed of surrogate salt and PBB3 brine (30 wt % H_2O), at 150 and 200°C, for time periods up to 6 months. At the end of the test no pitting or crevice corrosion tendency was observed on any specimen. The same type of specimens were exposed to two-phase salt-brine environments composed of surrogate salt and PBB3 brine, irradiated at cobalt-60 irradiation intensities of 1.1 to 2.5 krad/hour and held at 150°C for test durations of 6 months. No pitting corrosion or crevice corrosion was observed in these studies. The absence of nonuniform corrosion is consistent with electrochemical observations made on this high-Mg anoxic system.

The data base that currently exists describing the potential degradation of A216 steel by nonuniform corrosion contains only negative results, i.e., nonuniform corrosion has not been observed. The testing is not sufficient to define whether nonuniform corrosion could be association with long exposure times, low amounts of moisture (with high Mg^{2+} concentrations), minor salt constituents, microbial activity, prototypic fabrication variables, scale (size effects), or preclosure (air with brine) conditions.

Stress Corrosion Cracking. Stress corrosion cracking (SCC) is considered a potentially significant failure mode for mild steels in a repository environment, as chemical species (e.g., HS^-, NO_3^-, CO_3^{2-}, HCO_3^-, NH_4^+, Cl^-) are likely to be present that are known to be able to cause SCC in mild steels. The cracking agents may be concentrated by several mechanisms, and additional cracking agents may be generated by radiation. Also, radiation fields may move the free corrosion potential in the noble direction, which may increase the likelihood of SCC.

Work performed to determine the susceptibility of A216 steel to SCC utilized four experimental approaches:

- U-bend tests (static)
- Bolt-loaded fracture tests (static)
- Slow-strain-rate tests (SSR) (dynamic)
- Corrosion fatigue tests (dynamic).

The SCC testing performed using salt-repository-relevant anoxic brine environments and several static and dynamic test methods has revealed no positive susceptibility of A216 steel to SCC. The observation is consistent with the observed polarization behavior. The presently perceived uncertainties lie in defining the behavior of the specific prototypic container material under the heat treatment and weldment conditions that will be used in the actual containment manufacture; determining the effect of minor salt constituents, such as HS^- and $S^=$, and microbiological factors on SCC susceptibility; determining the effect of O_2, as in anticipated preclosure environmental conditions; and determining more precisely the threshold stress intensity separating the mechanical from the environmental aspects of crack propagation.

2.3.2.2 Titanium Alloys

Evaluation of the corrosion performance of Ti Grade 12 has consisted of anticipating its degradation modes under salt repository conditions by reviewing the relevant literature, by assessing the available degradation data, and by performing laboratory-scale experimental tests to determine Ti Grade 12 corrosion behavior in test environments specific to those expected in a salt repository.

The potential degradation modes of Ti Grade 12 have been reviewed (Westerman et al., 1987). The five degradation modes considered most significant for this alloy are presented below in their (approximate) order of perceived significance, from most severe to least severe:

- Nonuniform (pitting/crevice) corrosion
- Hydrogen-induced delayed failure
- Hydrogen embrittlement
- Stress corrosion cracking
- General (uniform) corrosion.

Pitting of Ti Grade 12 has been observed under unusual surface contamination conditions (iron embedment on Ti Grade 12 tubing), but ordinarily this material is resistant to pitting attack. Potentially much more serious is Ti Grade 12's tendency to exhibit crevice corrosion behavior in nonirradiated, elevated-temperature brines (Ahn and Soo, 1982a,b; Stahl and Miller, 1983) and irradiated, elevated-temperature brine (Westerman et al., 1987). Work conducted will be described briefly here, because of the extent of attack observed and the impact that the observation had on the candidacy of the alloy as a viable container material option. The test was intended to provide both crevice corrosion and hydrogen absorption data; only the former will be discussed in this subsection.

The material was exposed to the test environment in the form of coupons cut from 0.79-mm (0.032-in) sheet stock, in the as-received and in the SiC-grit-blasted condition. The test environment was anoxic high-Mg^{2+} (PBB3) brine, irradiated with cobalt-60 at an intensity of 2 x 10^4 rad/hour, and maintained at a temperature of 150°C.

At the conclusion of the test (a maximum exposure of 18 months for some specimens), all of the specimens were removed for analysis. All of the crevice pair specimens (six pairs, both as-received and grit-blasted) showed some degree of corrosion where they contacted each other. These crevice pair specimens had a total exposure of 10.5 months. A specimen exhibiting what appeared to be the most advanced corrosion was cleaned by light abrasion with emery paper. Several sites of advanced crevice corrosion attack existed under the TiO_2 layer. The attack had proceeded to an average depth of ~0.07 mm (0.003 in) in several places on the specimen.

The extent of crevice corrosion found during the course of this study indicated that this material can exhibit a potentially severe crevice corrosion problem under repository-simulating conditions. The study was not of sufficient duration or scope to enable a determination of the rate at which the attack would proceed over long test times or of whether the attack would stop through eventual passivation.

Hydrogen-induced delayed failure involves the growth of subcritical cracks under a static load at stress intensities less than K_{IC}, the critical stress intensity for crack propagation in a plane strain fracture mode. It is attributed to the formation of a hydride phase in the highly stressed region near the crack tip, with a subsequent fracture of the hydride and advancement of the crack. Such cracking is observed in the a + B class of titanium alloys of which Ti Grade 12 is a member. Information is not currently available that would allow an assessment of the subsceptibility of Ti Grade 12 to this failure mode. It is considered important, and its resolution is vital to the candidacy of the material as a waste package material.

Hydrogen embrittlement does not appear to be a severe problem with Ti Grade 12, as high hydrogen concentrations are required (greater than 400 ppm) before a significant reduction in ductility is observed in SSR tests (Sorenson and Ruppen, 1984). The data were limited in strain rate and method of hydrogen charging, however, and so must be considered only preliminary. Absorption of H_2 by Ti Grade 12 exposed to irradiated, high-Mg brine at 150°C appears to be an insignificant problem.

A number of investigations have examined the potential susceptibility of Ti Grade 12 to stress corrosion cracking and have found no cause for concern. The general corrosion performance of Ti Grade 12 under anticipated repository environment conditions is judged to be totally satisfactory.

2.3.2.3 Nickel Alloys

Because of the high general corrosion rates of the reference cast mild steel in high-Mg brines or salt-brine mixtures, and the potential difficulty of eliminating the possibility of finding large quantities of high-Mg brines in a salt repository, consideration was given to the selection of an alternate waste package container material for subsequent development and testing. As previously discussed, the titanium alloy Ti Grade 12 was found to have excellent resistance to general corrosion in repository-relevant brines, but has exhibited crevice corrosion in irradiation-corrosion tests and is potentially vulnerable to hydrogen-assisted cracking (Abrego and Rack, 1981; Sorenson and Ruppen, 1984). For these reasons Ti Grade 12 is not considered to be a completely acceptable alternate material at the present time.

Nickel-based alloys show the most promise of resisting the extremely hostile environments expected in a salt repository. Two alloys seem particularly well suited for container material service, based on the available literature: Inconel 625 and Hastelloy C-4.

The corrosion of Inconel 625 was found to have a negligible corrosion rate in WIPP Brine A (a high-magnesium brine) at 250°C. It exhibited very low corrosion rates in geothermal screening studies; the corrosion rate in deoxygenated 1% NaCl with 10 mg/l H_2S was 0.005 mm/yr at 250°C, and the rate in deoxygenated 20% NaCl was 0.003 mm/yr at the same temperature.

It is clear from the literature that Inconel 625 has good resistance to general corrosion, pitting, weld corrosion, and crevice corrosion in deoxygenated brines. The addition of oxygen increases the corrosion rate and the susceptibility to localized corrosion and cracking. For example, the corrosion rate of Inconel 625 in hypersaline geothermal brine at 232°C was negligible when the brine was deaerated, but increased to 0.49 mm/yr in brine containing 100 ppm dissolved oxygen. The susceptibility to stress corrosion cracking, pitting, and weld corrosion also increased. Brine entering a salt repository is expected to be anoxic; however, the effect of radiolysis products on the metal needs to be investigated.

Hastelloy C-4 is a low-carbon, Ti-stabilized variation on the Hastelloy C composition, which is alleged to exhibit a more stable, aging-resistant microstructure than Hastelloy C or C-276 through suppression of molybdenum carbide precipitation.

Hastelloy C-4 was studied to determine its resistance to general corrosion and localized corrosion in German waste package work. It was found to have low corrosion rates (0.02 to 0.04 um/yr) and good resistance to pitting and stress corrosion cracking in tests to two years in duration in unirradiated conditions, but was found to be susceptible to crevice corrosion under certain test conditions. The presence of high-intensity gamma irradiation (10^5 rad/h) increased the susceptibility to pitting and crevice corrosion. The German investigators concluded that Hastelloy C-4 would be suitable for use in a waste package under lower radiation intensity conditions.

Nickel-based alloys have been tested in many applications, and have been shown to have excellent corrosion resistance even in aggressive environments. Some of these alloys have been used extensively in autoclaves, test canisters, and thermocouple sheaths. For example, ~25 test canisters and autoclaves made of Inconel 600, Inconel 625, and Hastelloy C have been used for up to 7 years to contain brine or brine-salt mixtures, and no failures have been observed, although minor pitting occurred in one Inconel 600 autoclave exposed to oxic Brine "A" at 150°C, and one Inconel 600 sheathed thermocouple showed significant attack. (Both Inconel 625 and Hastelloy C-4 are more corrosion resistant, due to their high Mo contents, than Inconel 600, which is not considered a candidate waste package container material.)

Based on data from the literature and experience, it was recommended that both Inconel 625 and Hastelloy C-4 be evaluated for use as alternate materials. A test plan was developed but not yet initiated.

2.3.3 Packing Materials

Several different materials have been investigated as possible packing materials to surround the waste containers in a geologic repository. Desireable qualities of these materials, whether used individually or in combination, depend on what function they will be required to provide. Past research has emphasized clay for its sealant and water absorbent abilities. Various other materials including oxides, hydroxides, silicates, anhydrites, and cement-based mixtures are now being considered in a salt repository.

In the following sections, past choices for packing materials are listed. These materials are reviewed, along with new candidate materials, in light of the properties required to meet conditions in a salt repository.

2.3.3.1 Past Research on Packing Materials

A considerable number of materials have been investigated as possible packing materials, but most of the emphasis tended to be on the expandable clays. Some of these materials, along with comments on their properties, are listed below:

Material	Qualities
Clays (bentonite)	Good water and radionuclide sorbant, poor thermal conductivity and stability
Clay plus sand/rock	Good radionuclide and moderate water sorbant; clay may have poor chemical stability
Zeolites	Good selective sorbant, used as additive
Desiccants	Good water sorbant, but poor radionuclide sorbant and poor chemical stability
Metals	Considered mostly as additives to lower Eh
Cements	If used alone, possible excessive brittleness and porosity
Tailored mixes	Can be designed for specific properties

Clay and Clay-Based Materials. When considering inexpensive and easily obtained materials for preventing water ingress, swelling clays, such as the montmorillonite-rich bentonite, have invariably been the first choice. Bentonite has been used for many years for sealing various underground structures against water penetration. These clays, by virtue of the pressure they generate by swelling when in contact with water, tend to greatly restrict water flow. For example, Wheelright et al. (1981) estimated that hydraulic conductivities would be 10^{-12} cm/sec or less. The use of saturated clay as a packing material could greatly limit the amount of water reaching the waste package materials because the only way water could pass through the clay would be by diffusion.

The concept of using compressed clay as a packing material in deep rock repositories was probably first advanced by the Swedish KBS Program (e.g., Pusch, 1979). Temperature can decrease the swelling ability of bentonites and alter their mineralogy so in some applications, such as the Swedish KBS Program, limiting temperature to <100°C has been suggested. This temperature restriction may be an unacceptable restraint to current salt repository designs.

There is disagreement over the time required to saturate bentonite. Wheelright et al. (1981) estimated that saturation would take thousands of years at low hydraulic pressures. Later work (Westsik et al., 1983) indicated that saturation times may be much shorter, though the hydraulic conductivity estimates were in agreement. The disagreement in saturation times results from different assumptions about hydraulic gradients.

Another advantage of clays is that they would be field-emplaced to a high fraction of their theoretical density because of their plasticity. This compression would increase the clay's thermal conductivity and its ability to establish a thermal bond between the waste package and the host rock. Even when compressed, however, typical swelling clays have a fairly low thermal conductivity (0.6 W/m-K for bentonite compacted to 1.98 g/cm^2) (Moss and Molecke, 1983). One remedy is to add crushed mining spoils or quartz sand to the clay.

The chemical characteristics of ground water evolve through interaction with host rocks while migrating through pores and fractures. Clays are natural weathering products of crystalline aluminosilicate rock, so, in addition to properties of expansion for certain clay types, clay minerals tend to be chemically compatible with the host rocks and associated ground waters in aluminosilicate systems. Consequently they have been studied extensively as possible packing materials in potential tuff repository sites located above the water table. In the former case, ground-water flow is expected to be continuous and in the latter it will be intermittent, but the primary function of the clay materials, in these cases bentonite, is still to exclude water. In these aluminosilicate systems, it is probable that clay-based packing materials would remain chemically stable.

In the salt repository, however, the main evidence for long-term stability of clay in salt is based on clays that occur as depositional products in salt deposits. Some studies indicate that bentonite and bentonite-sand mixtures being considered for other repositories may not have the required stability. In addition to their thermal instability and low thermal conductivity, it is apparent that they may be chemically unstable in a brine environment. In short-term scoping studies, it was shown that bentonite can alter to illitic or Mg-bearing clays, which is likely to decrease the swelling ability of the clay. The effect of gamma radiation on clay stability is largely unknown but is expected to be minor under repository conditions.

Desiccating Oxides. Though bentonite has been the most extensively studied packing material, other materials have also been considered in rock repositories. Among these are desiccating oxides such as MgO, which would be capable of chemically reacting with water to keep it from corroding the waste container material (Simpson, 1980). These oxides would also function well in salt, but their effectiveness would be limited to the quantity of material that could be used without creating too much of a barrier to heat transfer. Also, desiccants would be successful only up to a given volume of nearby water supplied by the rock and, after hydration, would tend to become brittle. Additionally, desiccants may lose their effectiveness by reacting with moisture in the local environment during the filling and monitoring period before the repository is sealed.

Zeolites. Zeolites have also been considered, not for preventing water access to the waste package, but for sorbing radionuclides in solution after waste container failure and waste form dissolution. These materials tend to be selective, and the competition from high concentrations of cations in the brine may make them ineffective. Also, some zeolites may not be chemically stable under the near-field conditions. They may be more useful in mixtures as a specific sorbant.

Anhydrites. Another possible packing material is one with retrograde solubility, such as anhydrite ($CaSO_4$) or calcite ($CaCO_3$). Such a material would be more soluble at low temperature than at high. Consequently, incoming water would initially dissolve the material, but would deposit it nearer to the container because of the higher temperatures there. Eventually an impermeable solid structure might be built around the waste package which would effectively block the brine flow.

Tailored Mixtures. Some more elaborate schemes have also been suggested to tailor packing materials by adding certain materials to reduce oxygen activity or certain additives to react chemically with specific radionuclides such as the actinides. There is obviously a point of diminishing returns in terms of what can be done, at what cost, to a relatively minor part of the entire waste containment system. If it is cost-effective, however, including a tailored packing material in the waste package system may be desireable.

2.3.3.2 Packing Materials Recommended for Future Consideration

Packing materials used in a salt repository should ideally have the ability to remove water, lower Mg concentrations, and modify brine chemistry to promote precipitation and sorption of key radionuclides while retaining plasticity and thermal conductivity. These requirements may necessitate the development and investigation of mixtures of materials designed to optimize packing material functions. Several types of materials have properties that suggest they would be worth considering as packing materials or as components of packing mixtures. Some of these materials are considered in more detail below. Materials that have more promise are oxides and hydroxides, sodium silicates, and calcium silicates as well as mixtures based on these compounds.

Oxides and Hydroxides. These materials, particularly alkaline earth hydroxides, have been suggested as a means of removing Mg from brines because Mg hydroxide is less soluble than most of the other alkaline earth hydroxides.

Preliminary calculations suggest that Ca oxides and hydroxides might be among the better candidates, in part because they are readily available at low cost. These materials would also aid in neutralizing any acids that may be generated, and the available Ca might assist in the removal of sulfate through gypsum precipitation.

If used alone, some alkaline earth oxides tend to become brittle after hydration, and may produce very high pH values which would not be desireable for glass waste forms. The oxides tend to hydrate, even in air, and the dissolution of hydroxides is rapid. It may be better to consider these materials as components of mixtures where hydration and dissolution can be controlled. For example, the slower production of $Ca(OH)_2$ during the hydration of calcium silicate phases in cement materials might be more controllable and achieve the desired effect. If brine influx is relatively rapid, however, a large amount of available $Ca(OH)_2$ may be desirable.

Silicates. As an alternative method of lowering the Mg concentration of brines, it has been suggested that silica-based materials might be more effective than oxides and hydroxides. These materials are reactive with dissolved Mg and readily form hydrated magnesium silicate reaction products, so they are also capable of removing water. Using a pure form of reactive silica, such as amorphous silica or silica gel, would result in the formation of magnesium silicates, which tend to produce hydrogen ions and cause a decrease in pH. As this may not be desirable, water-soluble sodium silicate, which yields the same result but with higher pH, has been proposed. On the basis of simple mass balance calculations, the use of sodium silicate as a packing material seems reasonable. Silicates may also be incorporated in appropriate mixtures. The material is readily available at reasonable cost.

Cement Mixtures. Another type of system that can remove Mg and water, but might be more controllable over longer time periods, involves calcium silicates and reactions encountered during the setting of cements. Cements and concretes have already been investigated as materials for sealing shafts of salt repositories (Wakeley and Roy, 1983), and the ability of these materials to set in a brine environment has been demonstrated. Their direct use as packing materials would not be warranted because they can become brittle and porous. The utilization of cement-based mixed materials, however, shows considerable promise that might be worth investigating in more detail. It might be feasible to design a mixture that will remove water and Mg at a predetermined rate, have good thermal conductivity, and retain a degree of plasticity while offering possible sorption sites and a relatively high pH environment.

2.4 PERFORMANCE ASSESSMENT

The Waste Package Performance Assessment (WPPA) code operates as part of the near-field waste package subsystem as a one-dimensional, systems-level code to estimate the performance of the waste package for expected and unexpected conditions. The WPPA code has been designed as a group of subroutines that interact to estimate the condition of the waste package as a function of time. The primary WPPA subroutines are as follows:

- Radiation
- Mechanical
- Thermal
- Nuclide release
- Corrosion
- Mass transport.

This section discusses investigations regarding the development of the nuclide release subroutines.

2.4.1 Borosilicate Glass Dissolution Modeling

Mechanistic models have described the evolution of the solution chemistry as a function of reaction progress by using geochemical codes (Grambow, 1982; Grambow et al., 1986). However, present restrictions in the thermodynamic data base for many solid and solution species and the inability to perform computations on solutions of greater than 0.1 molal ionic strength preclude the utilization of these codes for modeling realistic salt repository environments. As a consequence, an approximation was derived to the reaction progress model that retains many of the features of the model without relying on geochemical codes. The general form of the mass balance (described by McGrail and Strachan, 1987) is

$$V\Phi \frac{dc}{dt} = \sum_{i=1}^{M} \Omega i + \sum_{r=1}^{N} \nu_r I_r \qquad (1)$$

where V = an assumed well-mixed control volume surrounding the waste package

c = concentration of silicon

Φ = volume fraction of liquid in the control volume

Ω = external source of sink for silicon

ν = stoichiometric coefficient

I = rate law

Appropriate rate laws and constants in Equation 1 to model the dissolution behavior of SRL-165 defense waste glass in PBB3 brine at a temperature of 90°C have been defined.

The effect of iron can be described as a sink term that prolongs the time a glass dissolves at a significantly higher rate than the final rate of reaction. The corrosion of the cast steel included at the start of the tests resulted in a continuous production of these reactive corrosion products. The mechanism proposed is that a ferrous silicate alteration product forms as a result of the simultaneous dissolution of both the glass and the A216 steel. The precipitation rate of this phase is dependent on time and on the available steel surface area. The silica mass balance model was used to predict the observed variation in the silica content of the reaction product.

2.4.2 Spent Fuel Dissolution Modeling

Spent fuel dissolution models are currently immature because mechanisms of dissolution are not completely understood. Experimental data suggest that release from spent fuel can be conceptually divided into three regions: release from the fuel-cladding gap, release from the grain boundaries, and release from the UO_2 matrix.

In recent modeling efforts, it has been common to assume that the gap inventory and at least a portion of the grain boundary inventory are instantly released from spent fuel upon breaching of the waste package. This "instant release fraction" (primarily Cs and I) has been shown to be approximately equal to the fractional release of stable xenon, a readily measurable parameter that is usually in the range 0.03 to 10 percent of total xenon inventory (Johnson et al., 1983; 1985). The stable xenon release can then be correlated with the linear power rating of the fuel to provide a predictive relationship between the instant release fraction and the irradiation history of the fuel.

The gradual decrease in the release rate of radionuclides from spent fuel after the instant release period is often attributed to a gradual depletion of the grain boundary inventory. Some of the available data appear to follow a $t^{-1/2}$ dependence, which has led some investigators to speculate that diffusion processes could be rate controlling.

The release rate of radionuclides from the UO_2 matrix is believed to depend on the dissolution rate of UO_2 (i.e., congruent release). Under oxidizing conditions, UO_2 is not a thermodynamically stable solid, and the dissolution of UO_2 is widely assumed to be preceded by sequential oxidation of UO_2, with the eventual formation of U(VI) in solution (Wang and Katayama, 1982; Thomas and Till, 1984; Johnson et al., 1982; Shoesmith et al., 1984). Under these circumstances, the solubility of U will be controlled by a stable U(VI)-bearing solid phase.

A kinetic model of Grandstaff (1986) appears to adequately predict the dissolution rate of uraninite (natural UO_2) under oxidizing conditions, but it is not directly applicable to salt repository systems. The model treats pH, dissolved oxygen concentration, carbonate concentration, and temperature (among other variables) as completely independent. These variables are not independent except possibly under conditions where the system is far from equilibrium. Also, the model is applicable only at pHs ranging from 4 to 6, and it is not valid when either the carbonate or the oxygen concentration approaches zero.

Salt repository brines are expected to be buffered to relatively low oxidation potentials by iron corrosion products (or perhaps by iron itself) from waste package container materials. Under these conditions, UO_2 is expected to be thermodynamically stable, and the solubility of uranium should be extremely low. Unfortunately, very few spent fuel dissolution experiments have been conducted under these conditions (most have been conducted under air-saturated conditions). Consequently, predictive modeling of UO_2 dissolution rates under reducing conditions is virtually nonexistent.

Radionuclide release rates have been measured in air-saturated salt brine, but only for relatively short periods of time (3 years). Thus, the possibility that spent fuel dissolution rates will eventually reach a steady-state value and thereafter will dissolve congruently has not been experimentally verified. This is an important question because congruent dissolution, if it can be shown to apply, together with a limited dissolution rate of the UO_2 matrix, could limit solution concentrations of highly soluble elements such as Cs and I to values significantly below their solubility limits.

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CHAPTER 3

SITE (WBS 3.0)

The purpose of this chapter is (1) to provide an overview of geotechnical, environmental, and socioeconomic activities resulting in the selection of the bedded salt site in Deaf Smith County, Texas, that was to have been characterized as a candidate site for the first nuclear repository, and (2) to provide a summary status of geotechnical, environmental, and socioeconomic plans to characterize the site.

Selection of the Deaf Smith County, Texas, salt site was the last phase of a multi-phase process in which successively smaller geographic territories--United States, regions, areas, locations, and sites--were screened for suitability for a repository. Each phase involved screening the geotechnical, environmental, and socioeconomic characteristics of a territory through a set of siting criteria or guidelines, each successive screening using progressively more detailed information on characteristics and on siting criteria or guidelines. The National Plan for Siting a High-Level Radioactive Waste Repository (DOE, 1982) provides a detailed discussion of this site selection process.

The initial screening, in the 1960s and 1970s, was at a national level of rock salt deposits or provinces in the United States. This national screening identified four salt regions for screening at greater detail.

On the basis of regional screening, the following areas were selected for geotechnical, environmental, and socioeconomic characterization:

- Areas in Gulf Coast Region:
  - North Louisiana: Rayburn's and Vacherie Domes
  - Southeast Mississippi: Cyprus Creek, Lampton, and Richton Domes
  - East Texas: Keechi and Oakwood Domes
- Areas in Paradox Basin Region:
  - Elk Ridge, Gibson Dome, Lisbon Valley, and Salt Valley
- Areas in Permian Basin Region:
  - Palo Duro and Dalhart Basins in Texas Panhandle

The DOE's area characterization studies resulted in identification, in 1983, of the following seven potential salt sites in the following locations:

- Cypress Creek and Richton Domes sites in southeast Mississippi
- Vacherie Dome site in north Louisiana

- Davis and Lavender Canyon sites in the Gibson Dome area of southeast Utah

- Two sites in the Palo Duro Basin, one in each of Deaf Smith and Swisher Counties in the Texas Panhandle.

The Secretary of the Department of Energy nominated three of the above salt sites as candidates for site characterization--the Richton Dome, Davis Canyon, and Deaf Smith County sites. As required by the Nuclear Waste Policy Act of 1982, an Environmental Assessment (EA) report was prepared for each of the three sites (DOE, 1986a,b,c). On the basis of comparative evaluation of the three sites (Chapter 7 of each EA), the Deaf Smith County, Texas, site was selected as a preferred site for characterization.

The screening process which resulted in the identification of three candidate salt sites for a repository is described in Sections 3.1 and 3.2. An overview of the geotechnical and environmental studies conducted to select the three sites is in Sections 3.3 and 3.4, respectively. Section 3.5 presents the titles and status of geotechnical and environmental plans that would have been implemented to characterize the site in Deaf Smith County, Texas.

3.1 SCREENING PROCESS

The steps in the screening process were as follows:

1. A <u>national</u> survey of a particular rock type (e.g., salt formations) is conducted.

2. The <u>national</u> survey data are used to identify <u>regions</u> of up to several states in extent that appear suitable for further investigation.

3. The results of the <u>regional</u> investigations are used to identify <u>areas</u> of from 1,000 to 10,000 square miles within the <u>regions</u> for more detailed investigations.

4. The <u>area</u> data are then used to identify specific <u>locations</u> of from tens to hundreds of square miles within the <u>area</u> for more intensive evaluation.

5. The <u>location</u> phase of the investigation results in the identification of a <u>site</u> or <u>sites</u> that appear to be most suitable for the construction of a nuclear waste repository.

3.1.1 <u>National Survey</u>

The national surveys relied on published geologic and environmental information to identify regions for further investigation. Information was collected on the depth and extent of geologic formations.

Salt was first recommended as a potentially suitable host rock for waste disposal in 1957, after the National Academy of Sciences-National Research Council evaluated many options (NAS-NRC, 1957). This recommendation was reaffirmed in subsequent reports (e.g., American Physical Society, 1978; NAS-NRC, 1970). Rock salt, which occurs both as bedded salt and in salt domes, was identified as having several characteristics that are favorable for isolating radioactive waste.

In 1958, the U.S. Geological Survey undertook a study for the Atomic Energy Commission to identify those salt deposits in the United States that might contain possible disposal sites (Pierce and Rich, 1962). Further screening of the entire United States in the 1960s and 1970s (Johnson and Gonzales, 1978) led to the identification of four large regions that are underlain by rock salt of sufficient depth and thickness to accommodate a repository and represent diverse hydrogeologic conditions. The four regions were as follows:

- Bedded salt in the Michigan and the Appalachian Basins of southern Michigan, northeastern Ohio, western Pennsylvania, and western New York (also called the "Salina Basin")

- Salt domes within a large part of the Gulf Coastal Plain in Texas, Louisiana, and Mississippi

- Bedded salt in the Permian Basin of southwestern Kansas, western Oklahoma, northwestern Texas, and eastern New Mexico

- Bedded salt in the Paradox Basin of southeastern Utah, southwestern Colorado, and northernmost Arizona and New Mexico.

This screening at the national level served as the basis for all subsequent screening in salt.

3.1.2 Regional Studies

Regional studies evaluated large areas of land identified during the national screening survey to obtain geologic and environmental information. These studies consisted almost entirely of a review of existing literature. This information was used to compile tables and maps for the selection of areas within the region for investigation. Information was collected on natural resources; population density and distribution; types and size of industries; types of ecosystems and their distribution; national landmarks and historical and archaeological sites; dedicated land uses such as national parks and wildlife areas; atmospheric conditions; and transportation networks. Many of these factors directly or indirectly related to the ability of DOE in the future to demonstrate compliance with environmental statutes and regulations.

3.1.3 Area Studies

After proceeding to the area phase, further screening of the salt deposits in the Salina Basin was deferred. The studies of the Salina region were not specific enough to judge that any part of the region was suitable or unsuitable for a repository.

The same factors as in the regional studies were evaluated for the area studies but within a smaller area and in much greater detail.

Geologic field work, initiated during this phase, included drilling deep boreholes through possible repository formations and occasionally down to basement rock. Each of the boreholes involved 4 to 10 acres of land disturbance and consequently potential impacts to historic and archaeological resources and other resources and endangered species. In accordance with applicable regulations, prior to the initiation of these ground-disturbing activities the Department of Energy prepared Environmental Evaluations for each site.

The environmental evaluations involved onsite archaeological and ecosystem investigations and communication concerning the results of these surveys with appropriate Federal and State officials.

Except for these environmental evaluations, environmental studies during the area phase continued to be based primarily on literature surveys and consultation with agencies, local experts, and institutions such as colleges and universities. The scope of these studies included an evaluation of surface- and ground-water relationships, characteristics, and use; a general characterization of the climate; an investigation of the occurrence of extreme meteorological events; an analysis of population density, location of urban centers, major transportation networks, land use, and industrial and agricultural activities; identification of potentially conflicting sites; and a consideration of natural and manmade biological systems.

The environmental studies were all designed to contribute to the selection of a site which would meet all environmental regulatory and statutory requirements, be licensable through the NRC review process, and fulfill the intent of the National Environmental Policy Act.

3.1.4 Location Studies

Location studies represented a further focusing of the data collection effort on land areas of tens to hundreds of square miles. The primary purpose of location studies was to select one or more sites that appeared most suitable for siting a nuclear waste repository.

Data gathering during this step included more extensive drilling to obtain geologic and hydrologic information; geophysical evaluation; environmental field work; aerial photography; and extensive contact with local, State, and other Federal agencies, local and regional experts, and universities. The level of information gathered had to be sufficient to identify a site and defend the site selection process, initiate site specific engineering design, obtain necessary construction permits and investigation-related permits for the next phase of study, and prepare an environmental analysis (i.e., an environmental assessment or environmental impact statement) of site characterization activities. This was the period when the majority of the data base for the subsequent Environmental Assessments was established (Section 5.2.2).

3.1.5 Site Characterization

Detailed site characterization was to be the final step in the process of siting a nuclear waste repository. Baseline studies of a year or more of background radiation, meteorology, air quality, and local ecosystems were proposed. Detailed socioeconomic surveys of the affected communities and nearby counties were to be initiated in order, through the use of socioeconomic models, to predict community and economic impacts. Onsite meteorological data were to be gathered to predict air quality and radiological effects. Potential radiological pathways were to be determined and a demographic distribution within 50 miles (80 kilometers) of the site was to be developed.

3.2 SCREENING OF THE THREE SELECTED REGIONS

3.2.1 Screening of the Gulf Coast Salt Domes Region

More than 500 salt domes are in the Gulf Coast salt-dome basin of Texas, Louisiana, Mississippi, and areas offshore from these States. An initial screening by the U.S. Geological Survey (USGS) eliminated all offshore domes because siting a repository under water would probably not be feasible. The application of this criterion eliminated about half the domes. The USGS also evaluated the remaining 263 onshore domes (i.e., Gulf interior domes) and identified 36 as being potentially acceptable for a repository and another 89 that were worthy of further study (Anderson et al., 1973). The USGS screening factors were the depth to the top of the dome, present use for gas storage, and hydrocarbon production.

The DOE and its predecessor agencies conducted regional studies of the 125 salt domes identified in the above-mentioned USGS screening. All but 11 of the domes were eliminated on the basis of three screening factors: the depth to the salt, the lateral extent of the dome, and the history of use for hydrocarbon production or storage (BNI and LETCO, 1980). Three of the 11 domes were removed from consideration on the basis of environmental factors, and a fourth was eliminated because solution mining at the site contributed to a collapse of strata above the dome.

Area characterization studies were completed for the seven remaining dome areas: Rayburn's and Vacherie Domes in Louisiana; Cypress Creek, Lampton, and Richton Domes in Mississippi; and Keechi and Oakwood Domes in Texas. The geologic fieldwork conducted during this phase included the drilling of deep holes to collect rock cores from the aquifers and other strata for laboratory tests of their properties, and geophysical surveys to determine the underlying rock structures. The area environmental studies included descriptions of the plant and animal communities, surface- and ground-water systems, weather conditions, land use, and socioeconomic characteristics. An evaluation of the seven domes on the basis of the DOE's criteria is summarized in a location recommendation report (ONWI, 1982a).

In the area characterization studies, the DOE chose a repository siting criterion that was more restrictive than the one used in the regional screening studies. The application of this stricter criterion resulted in the elimination of Keechi, Rayburn's, and Lampton Domes (ONWI, 1982a). Thus, at the conclusion of area characterization, the Vacherie, Richton, Oakwood, and Cypress Creek Domes were recommended for further screening. After further review of the area characterization studies, the Oakwood Dome was deferred from further consideration because of uncertainties raised by large-scale petroleum exploration.

The DOE identified the Cypress Creek, Richton, and Vacherie Domes as potentially acceptable sites in February 1983. Information on the three dome sites is in the Environmental Assessments (EAs) (DOE, 1984a,b; DOE, 1986c).

3.2.2 Screening of the Paradox Basin Bedded Salt Region

Screening criteria were developed for the bedded salt of the Paradox Basin, which the USGS had identified as worthy of further investigation (Pierce and Rich, 1962). The following factors were applied to identify areas for further investigation (Brunton and McClain, 1977; DOE, 1981): the depth to, and the thickness of, the salt; mapped faults; surface igneous features; hydrocarbon and mineral resources; and potential for flooding. The results of this screening were integrated with the results of screening for environmental and socioeconomic factors, such as proximity to urban area and the presence of certain dedicated lands. On the basis of this regional screening, four areas were recommended for further study: Gibson Dome, Elk Ridge, Lisbon Valley, and Salt Valley (BGI and WGG, 1982a).

The primary screening factors used to identify potentially favorable locations within the four areas were the depth to the salt, the thickness of the salt, proximity to faults and boreholes, and proximity to the boundaries of dedicated lands (ONWI, 1982b). These screening factors were judged to have the strongest potential for differentiating possible locations within the areas.

Salt Valley and Lisbon Valley were both deferred from further consideration. Salt Valley is underlain by a structurally complex anticline. Areas with adequate depth to the salt in Lisbon Valley were too close to zones of mapped surface faults; also, numerous boreholes penetrate the potential repository salt bed (BGI and WCC, 1982b).

Application of the screening factors to the Gibson Dome area resulted in a location of 57 square miles near the center of the area that contained appropriately deep and thick salt deposits and was sufficiently far from faults or exploration boreholes that would make a site unsuitable. It was also outside the boundaries of the Canyonlands National Park. This location is referred to as the Gibson Dome location. The Elk Ridge area contained one location of about 6 square miles and several smaller ones, each less than 3 square miles, that met the screening criteria (ONWI, 1982b). The smaller locations were not large enough for a repository and were therefore excluded from further consideration. The larger location was designated the Elk Ridge location.

Further comparisons of the Gibson Dome and the Elk Ridge locations were made on the basis of more refined criteria that discriminated between them. The thickness of the salt, the thickness of the shale above and below the depth of a repository, and the minimum distance to salt-dissolution features were considered the most critical geologic discriminators. Archaeological sensitivity and site accessibility were considered the most important environmental factors. The Gibson Dome location was judged to be superior to the Elk Ridge location in terms of the number and relative importance of favorable factors and was selected as the preferred location (ONWI, 1982b).

During 1982 and 1983 three sites in the Gibson Dome location were identified for further evaluation: Davis Canyon, Lavender Canyon, and Harts Draw. Since much of the intrinsic value of southeastern Utah stems from its scenic and aesthetic character, a study of visual aesthetics was performed to evaluate the three sites (BGI, 1984b). Harts Draw was found to be less desirable than the sites at Davis Canyon and Lavender Canyon because it affords a greater total area of visibility, and it was eliminated from further consideration. In February 1983, Davis Canyon and Lavender Canyon were identified as potentially acceptable sites.

Information on the Davis Canyon and Lavender Canyon sites can be found in the Environmental Assessments (DOE, 1984c; 1986b).

3.2.3 Screening of the Permian Basin Bedded Salt Region

In 1976, the Permian bedded-salt deposits in the Texas Panhandle and western Oklahoma that had been identified in the USGS study (Pierce and Rich, 1962) were evaluated to determine whether they contained any areas that might be suitable for waste disposal (Johnson, 1976). This screening focused on five subbasins: the Anadarko, Palo Duro, Dalhart, Midland, and Delaware Basins. The primary screening factors were the depth to and the thickness of the salt, faults, seismic activity, salt dissolution, boreholes, underground mines, proximity to aquifers, mineral resources, and conflicting land uses, such as historical sites and State or national parks. All the subbasins contain salt beds of adequate thickness and depth. The Palo Duro and the Dalhart Basins had far less potential for oil and gas production and have not been penetrated as extensively by drilling as have the Anadarko, the Delaware, and the Midland Basins. Therefore, the Palo Duro and the Dalhart Basins were judged to be preferable to the other three and were recommended for further studies at the area stage (ONWI, 1983a). These two basins rated higher on six major screening factors: the depth to and the thickness of the salt, seismicity, known oil and gas deposits, the presence of exploratory boreholes, and evidence of salt dissolution.

More detailed geologic and environmental studies of the Palo Duro and the Dalhart Basins began in 1977, and screening criteria were developed to define locations with favorable characteristics. The screening criteria that were most useful in the area-to-location screening were the following: salt depth and thickness, salt purity, existing and abandoned oil and gas fields, flooding, urban areas, and conflicting land use. Six locations in parts of Deaf Smith, Swisher, Oldham, Briscoe, Armstrong, Randall, and Potter Counties, Texas, met the screening criteria. A second set of criteria was then applied to further differentiate among the six locations: distance from the margins of the Southern High Plains, distance from known oil and gas fields, more than one potential repository horizon, depth of salt, number of boreholes that penetrate the repository horizon, a large geographic area, low population densities, and potential land-use conflicts. After applying these criteria, the DOE decided to focus on the two locations that had the greatest likelihood of containing a suitable site, one in the northeastern Deaf Smith and southeastern Oldham Counties, and one in north-central Swisher County. All other locations in the Palo Duro Basin were deferred from further consideration (ONWI, 1983b). In February 1983, the DOE identified parts of Deaf Smith County and Swisher County as potentially acceptable sites and subsequently narrowed the size of the two sites to be considered at each location to 9 square miles each. Information on the two sites is in the EAs (DOE, 1984d; 1986a).

3.3 GEOTECHNICAL STUDIES SUPPORTING SCREENING

The following is a brief discussion of geotechnical studies performed as part of the Salt Repository Project. The discussion is subdivided into the three regions investigated: the Texas Panhandle portion of the Permian Basin; the Paradox Basin of southeastern Utah; and Gulf Coast Salt Domes in Texas, Louisiana, and Mississippi. Specific activities covered include drilling and sampling, including quantities of core recovered; analyses performed, including geomechanical, geochemical, and hydrologic; microearthquake monitoring; and storage of samples as of March 1988.

3.3.1 Permian Basin

Unlike the Paradox Basin and Gulf Coast Salt Domes, where activities ceased for the most part when the Deaf Smith County site was selected as the single salt site, an arbitrary division is required in discussing Permian Basin work. Work conducted prior to Presidential approval of the Deaf Smith County site is covered in this section. Activities planned during site characterization are in Section 3.4.

3.3.1.1 Drilling and Sampling

A total of 13 boreholes were drilled in the Texas Permian Basin exclusively for this project. In 1978, the wells, DOE-Gruy Federal No. 1 Rex White in Randall County and DOE Gruy Federal No. 1 Grabbe in Swisher County, were cored, geophysically logged, sampled, and hydraulically tested (BEG, 1984a,b). Between June and October 1981, the Stone & Webster Engineering Corporation (SWEC) Sawyer No. 1 (PD-3) in Donley County was drilled through the entire Permian sequence to a depth of 4,806 feet with 4-inch core recovered from the majority of the interval between 350 and 3,938 feet (SWEC, 1984a). The SWEC Mansfield No. 1 (PD-4) in Oldham County was drilled to a depth of 5,180 feet during the period October 1981 to March 1982; continuous coring was performed in the intervals 46 to 3,540 feet, 4,022 to 4,213 feet, and 4,393 to 4,995 feet (SWEC, 1984b).

The SWEC G. Friemel No. 1 (PD-5) in Deaf Smith County was drilled to a depth of 2,710 feet (into the Lower San Andres Unit 3 formation) during March 1982, with essentially continuous 4-inch core recovered from depths of 1,190 to 1,312 feet and 1,709 to 2,710 feet (SWEC, 1984c). The SWEC Detten No. 1 (PD-6) in Deaf Smith County, drilled between February and May 1982, was drilled to a depth of 2,839 feet (into the Lower San Andres Unit 4 basal carbonate); essentially continuous 4-inch core was recovered from the intervals 1,129 to 1,423 feet and 1,885 to 2,839 feet (SWEC, 1984d). The well SWEC Zeeck No. 1 (PD-7) in Swisher County was drilled to a depth of 7,641 feet between April and August 1982. A total of 1,993 feet of 4-inch core was recovered from five separate intervals (SWEC, 1984e). The SWEC Harmon No. 1 (PD-8) in Swisher County was drilled to a depth of 3,052 feet (in the Lower San Andres Unit 2) between July and September 1982. Core was recovered from the two intervals 1,070 to 1,303 feet and 1,804 to 3,052 feet, a total of 1,481 feet (SWEC, 1984f).

The SWEC J. Friemel No. 1 (PD-9), closest of the 13 wells to the Deaf Smith County site, was drilled to a depth of 8,283 feet (into the Precambrian basement) between October 1982 and March 1983. A total of 2,782 feet of 4-inch core was recovered from seven intervals (SWEC, 1984g). The SWEC Holtzclaw No. 1 (PD-10) in Randall County was drilled to a depth of 2,878 feet (Lower San Andres, Unit 3) during the period March to April 1983. A total of 902 feet of 4-inch diameter core was recovered from the two intervals 1,079 to 1,401 feet and 2,304 to 2,884 feet (SWEC, 1984h).

The final three holes were unique in that they were designed specifically to investigate the salt dissolution zone and were not cored. The SWEC Sawyer No. 2, Mansfield No. 2, and Detten No. 2 were drilled 100 to 200 feet from the corresponding No. 1 wells in late 1982 and early 1983 (SWEC, 1984i).

Additional fluid samples and hydraulic data were obtained from existing wells throughout the Permian Basin. Results are provided in Section 3.3.2.1 of the Environmental Assessments (DOE, 1984d, 1986a).

3.3.1.2 Analyses

Geomechanical tests from Permian Basin samples consist of index tests, triaxial compression tests, uniaxial (unconfined) compression tests, and laboratory creep tests. Index tests (hardness, slake, durability, density, porosity, and water content) were performed on core from the Mansfield No. 1, Detten No. 1, G. Friemel No. 1, and J. Friemel No. 1 wells (SWEC, 1984j,k,l; 1985). Unconfined compression and indirect tensile tests were performed on core from the same four wells (SWEC, 1984k,m,n,o; Pfeifle et al., 1983). Point-load tests were performed at approximately 2-meter intervals on core from the J. Friemel No. 1 well (SWEC, 1985). Triaxial compression tests at various confining pressures, including tests at elevated temperatures, were performed on salt and nonsalt samples. Creep tests were performed on core from the Deaf Smith wells (Pfeifle et al., 1983; SWEC, 1984n; Senseny et al., 1985; Senseny et al., 1986). Details of the tests are discussed in Section 3.2.6 of the Environmental Assessments (DOE, 1984d; 1986a).

Geochemical analyses consisted of mineralogy and petrology (numerous stratigraphic units), and fluid content and chemistry from the upper and lower aquifers and the Lower San Andres Unit 4. Mineralogical studies focused on mineral percentages and halite petrology (Hvorka et al., 1985) and clay analyses (Fukui, 1984a,b; Fukui and Hopping, 1984, 1985; Fukui and Dayvault, 1985; Fukui and Ealey, 1986; Hvorka, 1984; Hvorka et al., 1985; Hall and Weaver, 1985). Fluid samples from fresh-water aquifers were analyzed (Nativ and Smith, 1985), the deep-basin-brine aquifers (Bassett and Bentley, 1983; Fisher and Kreitler, 1984; Knauth and Beeunas, 1985; Means and Hubbard, 1985; Laul et al., 1985), and halite, mudstones, and fluid inclusions from the Lower San Andres Unit 4 (Fisher, 1984; Roedder 1982; 1984). Details are provided in the Environmental Assessments (DOE, 1984d; 1986a).

Information regarding permeability, porosity, saturated thickness, and flow rates were collected for the upper and lower aquifer systems and for selected intervals in the Permian aquitard. Permeabilities have been derived from drill stem tests, pump tests, repeat formation tests, and core analyses (SWEC, 1984h,i,j,k,l,m,n,o; 1985).

3.3.1.3 Microearthquake Monitoring

A 16-station seismographic network was installed in the Texas Panhandle during 1983 and 1984. Fourteen stations became operational on April 1, 1984, and the remaining two on October 1, 1984. The operation of this network is described in Acharya (1984) and Acharya and Tyrala (1986). Results of the monitoring program are discussed in Section 1.4.1.1 of the Salt Repository Project Site Characterization Plan (DOE, 1988) and in Section 3.2.5.3 of the Environmental Assessments (DOE, 1984d; 1986a).

3.3.1.4 Sample Storage

The Texas Bureau of Economic Geology was tasked with control of samples from the Permian Basin. As of March 1988, all samples collected were either in the possession of the Bureau's Well Sample and Core Library or were to be shipped by project contractors to the Bureau. Current estimates place the linear footage of 4-inch Palo Duro Basin core in the Bureau's possession in excess of 20,000 feet. Generally, the core has been sliced longitudinally into thirds, with one-third being put into a permanent rock core library and the remainder reserved for testing.

3.3.2 Paradox Basin

3.3.2.1 Drilling and Sampling

In the Paradox Basin, 11 project boreholes were drilled and one existing hole reopened and deepened. The borehole, Gibson Dome No. 1 (GD-1), located near both the Davis Canyon and Lavender Canyon sites, was drilled between March 1980 and January 1981, to a depth of 6,384 feet. The hole was continuously cored at 4-inch diameter, sampled, geophysically logged, hydrologically tested, and hydrofracture tested to determine in situ stresses (WCC, 1982a; Thackston et al., 1984).

The borehole, Elk Ridge No. 1 (ER-1), drilled between August and October 1981, to a depth of 3,469 feet, is located in the southwest corner of the Paradox Basin in the Elk Ridge study area. The hole was cored, sampled, geophysically logged, and subjected to a series of constant-rate pump-out tests (WCC, 1982b; Thackston et al., 1984). The existing E. J. Kubat borehole, located approximately 1 kilometer ENE of ER-1, was opened and deepened, geophysically logged, and hydraulically tested via a series of injection/recovery and slug tests. Sidewall cores were obtained at intervals throughout the Paradox Formation and the well was swabbed to obtain formation fluid samples (WCC, 1982c).

Nine holes were drilled in the salt anticline of the Salt Valley area. DOE Salt Valley Nos. 1-3, drilled in 1978, were at the corners of an approximately equilateral triangle. Salt Valley Nos. 1 and 2 were drilled to a depth of about 1,250 feet. Total depth of Salt Valley No. 3 was 4,074 feet with most of the section below surface pipe cored with a 4-inch bit. The holes were geophysically logged and hydrologically tested and sampled using drill stem testing methods (WCC, 1979; Rush et al., 1980). DOE Salt Valley Nos. 4-9 were drilled to depths of about 500 feet and the caprock of the Salt Valley anticline pump-tested. Samples of formation fluids were obtained and chemically analyzed (Wollitz et al., 1982).

Additional fluid samples and hydraulic data were obtained from existing wells throughout the Gibson Dome area. Results are provided in Section 3.3.2.1 of the Environmental Assessments (DOE, 1984c; 1986b).

3.3.2.2 Analyses

Thackston et al. (1984) describe results of hydrological testing. Results of hydrofracture testing of GD-1 are given in Nelson et al., (1982). Standard laboratory tests were used to determine the mechanical properties of salt and nonsalt strata. These tests included triaxial compression, uniaxial (unconfined) compression, creep, and Brazilian (indirect tension) tests (Pfeifle et al., 1983; Nelson et al., 1982). Laboratory tests also measured rock index properties, including point-load strength, abrasion resistance, and Schmidt Hammer rebound hardness (WCC, 1982c). All tests were conducted on samples from the GD-1 borehole. Results are also documented in Section 3.2.6 of the Environmental Assessment (DOE, 1986b).

Geochemical analyses consisted of mineralogy and petrology, brine content, and brine chemistry for samples from the GD-1 borehole. Core samples were characterized for weight percent mineralogy, chemistry, and brine content in halite, clay, and carnallite (Hite, 1983). Chemical compositions of fluid samples included organic content (WCC, 1982d), and were used to calculate Eh (McCulley et al., 1984). Results are documented in Section 3.2.7 of the Environmental Assessments (DOE, 1984c; 1986b).

3.3.2.3 Microearthquake Monitoring

Monitoring of microearthquakes in the Utah portion of the Paradox Basin was initiated in July 1979. The program consisted of three phases: a 6-month, 24-station network; two long-term subnetworks totaling 12 seismographic stations selected from the 24-station network; and a 6-month, 12-station network. Details are provided in Section 6.3 of Vol. I of the Geologic Characterization Report for the Paradox Basin Study Region--Utah Study Areas (WCC, 1982e-i; Wong, 1984) and in Section 3.2.5.2 of the Environmental Assessments (DOE, 1984c; 1986b).

3.3.2.4 Sample Storage

Paradox Basin samples were shipped to the Texas Bureau of Economic Geology upon the termination of the Salt Repository Project. Total core recovered during drilling was approximately 11,500 feet.

3.3.3 <u>Gulf Coast Salt Domes</u>

3.3.3.1 Drilling and Sampling

Law Engineering Testing Company (LETCO) in 1979 and 1980 drilled numerous boreholes to acquire geological and hydrological data and for geophysical logging over and near salt domes in Mississippi, north Louisiana, and east Texas during the area characterization phase. Information on these boreholes is given in the August 1982 ONWI reports prepared by LETCO and listed below.

North Louisiana: ONWI 181-185
Mississippi: ONWI 165, 167, 170-180
East Texas: ONWI 186-188

The ONWI-165 report pertains to 50 shallow (20 to 260 feet) borings in the Cypress Creek Salt Dome area of Mississippi, and the ONWI-167 report pertains to 35 shallow (12 to 550 feet) borings in the Richton Dome area, also in Mississippi. The number of boreholes described in each of the other reports ranges from one to five.

3.3.3.2 Analyses

Cypress Creek Dome

Geomechanical properties for overburden, caprock, and salt at the Cypress Creek Dome were not measured, but are estimated in Section 3.2.6.1 of the Environmental Assessment (DOE, 1984a). Thermal properties, including coefficient of linear thermal expansion, thermal conductivity, and specific heat, were measured on four intact salt and caprock core samples from MCCG-1 (Lagedrost and Capps, 1983). For specific results, see Section 3.2.6.2 of the Environmental Assessment (DOE, 1984a).

Mineralogic and chemical analyses were performed on salt, caprock, and sediments adjacent to the dome. Chemistry of ground water in sediments adjacent to the dome was also determined on a small number of samples. X-ray, scanning electron microscopy, and chemical studies were also conducted (LETCO, 1983). Details of analyses are discussed in Section 3.2.7 of the Environmental Assessment (DOE, 1984a).

Based on limited data, horizontal and vertical hydraulic conductivities and transmissivities have been determined for the six principal hydrogeologic units. Results are provided in Section 3.3.2 of the Environmental Assessment (DOE, 1984a).

Richton Dome

Geomechanical properties for caprock and salt at the Richton Dome were assessed using test specimens from borehole MRIG-9; typical tests included unconfirmed compression and indirect tension (Brazilian disc) (Pfeifle et al., 1983). Mechanical properties for overburden were estimated. Thermal properties, specifically coefficient of linear thermal expansion, thermal conductivity, and specific heat, were measured on four relatively intact caprock and salt-core samples from the MRIG-9 borehole (Lagedrost and Capps, 1983). Details are provided in Section 3.2.6 of the Environmental Assessment (DOE, 1986c).

Mineralogic and chemical analyses were performed on salt and caprock samples from drillhole MRIG-9 (Werner, 1985; Drumheller et al., 1982) and on sediments adjacent to the dome. Chemical analyses exist for samples of sediments adjacent to the dome, taken from drill holes MRIG-10 and MRIG-11, and for caprock samples from well MRIG-9 (Bentley, 1983). Details of analyses are provided in Section 3.2.7 of the Environmental Assessment (DOE, 1986c).

Horizontal and vertical hydraulic conductivities and transmissivities were determined for six principal hydrogeologic units. Results are discussed in detail in Section 3.3.2 of the Environmental Assessment (DOE, 1986c).

Vacherie Dome

Geomechanical tests, specifically uniaxial (unconfined) compression, indirect (Brazilian) tension, and triaxial constant stress-rate (creep) were conducted on samples from borehole DOE-Smith No. 1 (Pfeifle et al., 1983); geomechanical properties of the overburden were estimated. Thermal properties were measured on caprock and salt core samples from borehole DOE-Smith No. 1 (Lagedrost and Capps, 1983). Results are provided in Section 3.2.6 of the Environmental Assessment (DOE, 1984b).

Mineralogies were established for salt, caprock, and sediments adjacent to the dome. The hydrochemistry in the seven hydrogeologic units adjacent to the dome was determined. While caprock fluids were not measured, content and composition of water in the domal salt were determined. Results are discussed in Section 3.2.7 of the Environmental Assessment (DOE, 1984b).

Horizontal and vertical hydraulic conductivities and transmissivities were measured for the seven hydrogeologic units adjacent to the dome. Porosity data is limited. The reader is referred to Table 3-18 and Section 3.3.2 in the Environmental Assessment (DOE, 1984b) for a summary of the properties.

3.3.3.3 Microearthquake Monitoring

No earthquake monitoring network was installed in the Gulf Coast area.

3.3.3.4 Sample Storage

Gulf Coast samples were shipped to the Texas Bureau of Economic Geology upon termination of the Salt Repository Project.

Decommissioning and reclamation of field sites are planned to be accomplished in accordance with activity plans prepared by Stone & Webster Engineering Corporation. The plans are entitled as follows:

- Activity Plan for Decommissioning Boreholes and Wellsite Restoration in the Paradox Basin, Utah
- Activity Plan for Dismantling of Microearthquake Network
- Activity Plan for Reclamation Work in Louisiana
- Activity Plan for Well Plugging and Site Restoration of Test Hole Sites in Mississippi.

Activities associated with decommissioning and reclamation are categorized into (1) Planning and Access, (2) Subcontracting and Field Work, and (3) Documentation. It is expected that most Planning and Access activities will be completed by mid-May 1988. Field work was planned to begin in mid-May 1988. Because access is essential to maintaining a scheduled September 30, 1988, completion, that date may slip if access is postponed or denied. In any event, all decommissioning/reclamation activities will be done in accordance with activity plans prepared by SWEC and approved by DOE-HQ.

3.4 ENVIRONMENTAL AND SOCIOECONOMIC STUDIES SUPPORTING SCREENING

The Salt Program conducted numerous environmental and socioeconomic studies in response to legislative mandates (Public Law 97-425) and program objectives. Meeting these mandates and objectives, which called for protection of the environment and incorporation of environmental and socioeconomic factors in the DOE decision process, required that paper studies, field studies, and methods development be performed. Most of the studies discussed herein are described or referenced in the draft and final Environmental Assessments (EAs) (DOE, 1984a-d; DOE, 1986a-c). Additionally, data were obtained and evaluations were made to assess project compliance with Federal and State environmental laws and regulations (DOE, 1986a-c, Chapter 6).

The reports of studies performed are listed in Tables 3-1, 3-2, and 3-3 along with the decision step each supported. The environmental and socioeconomic study efforts of the program are discussed below, and major reports are listed in the reference list for this chapter. Additional summary-level detail can be found in final project reports of the Regulatory Project

Table 3-1. Documentation Supporting the Permian Basin Screening Process

Site Selection Surveys	Identified Land Units	Characterization Documentation
Nation-to-Region	Permian Basin Midland, Delaware, Anadarko, Palo Duro, and Dalhart Basins	Johnson, 1976, Y/OWI/SUB-4494/1 NUS, 1983a, ONWI-27 Johnson, 1978, Y/OWI/SUB-7414/1 [a]NUS, 1979a, b, c
Region-to-Area	Palo Duro, Dalhart Basins	NUS, 1983b, ONWI-28 SWEC, 1983, DOE/CH/10140-1 NUS, 1982a,b, ONWI-102(1)(2)
Area-to-Location	Palo Duro A Palo Duro B	ONWI, 1983b, DOE/CH/10140-2
Potentially Acceptable Sites	Deaf Smith Site Swisher Site	NUS, 1984j, BMI/ONWI-561 NUS, 1986e, 1986f NUS, 1986a-d, BMI/ONWI-620,-629,-617,-618 NUS, 1984e,f,h,k, ONWI-461,-526,-527,-528 DOE, 1984e,f,g, DOE/CH-10(1)(2)(3) NUS, 1984a,b,c,d,g,i, BMI/ONWI-508,-509, -510,-558,-559,-560 NUS, 1985a,b,c, BMI/ONWI-557,-574,-575
Nominated Site	Deaf Smith Site	DOE, 1986a BCD, 1986b, 1986e ONWI, 1986e
Site Characterization	Deaf Smith Site	See Discussion on Study Plan

a. Salina Basin work performed by NUS and ONWI.

Table 3-2. Documentation Supporting the Paradox Basin Screening Process

Site Selection Survey	Identified Land Units	Characterization Documentation
Nation-to-Region	Paradox Basin	BNI, 1980b, ONWI-68 WCC, 1983, ONWI-92
Region-to-Area	Gibson Dome, Elk Ridge Salt Valley, Lisbon Valley	BGI and WCC, 1982b, ONWI-291 WCC, 1982e-h, ONWI-290 BGI, 1982a, ONWI-144
Area-to-Location	Gibson Dome	BGI and WCC, 1982a, ONWI-36 BGI, 1982b, ONWI-404 BLM, 1982a-c
Potentially Acceptable Sites	Davis Canyon, Lavender Canyon	BGI, 1983a-c,f-i, ONWI-453,-454,-460,-468, -469,-470,-476, and -477 BGI, 1984a,b, ONWI-454, -471
Nominated Site	Davis Canyon	DOE, 1986b BCD, 1986a,d BGI, 1984a BNI, 1986a,b ONWI, 1986d BLM, 1985 Shaver, 1985

Table 3-3. Documentation Supporting the Gulf Coast Salt Dome Basin Screening Process

Site Selection Surveys	Identified Land Units	Characterization Documentation
Nation-to-Region	Gulf Coast Salt Dome Basin	Johnson, and Gonzales, 1978
Region-to-Area	Gulf Coastal Plain (Mississippi Salt Basin) and Basins in Texas, Louisiana, and Mississippi	BNI and LETCo, 1980, ONWI-18 BNI, 1980a, ONWI-67
Area-to-Location	Rayburn's, Vacherie, Cypress Creek, Lampton, Richton, Keechi, and Oakwood Domes	LETCo, 1981, ONWI-106 LETCo, 1982a, ONWI-117, LETCo, 1982b, ONWI-118, LETCo, 1982c, ONWI-119, LETCo, 1982d, ONWI-120, ONWI, 1982a, ONWI-109 BNI, 1982a,b,c, ONWI-192,-193,-194
Potentially Acceptable Sites	Cypress Creek, Richton, Vacherie Domes	ONWI, 1982a, ONWI-109
Nominated Site	Richton Dome	DOE, 1986c BCD, 1986c, 1986f BNI, 1984c, ONWI-499 BNI, 1986d, 1986e ONWI, 1986f

Major peer and technical reviews were performed on the air quality modeling work.

3.4.2 Acoustics

Field measurements were made during the Permian and Paradox location study phases (NUS, 1984h, BMI/ONWI-526, -528; BGI, 1983g, ONWI-460). Field measurements of ambient noise levels in Canyonlands National Park were obtained from others who made the measurements for the National Park Service. Existing data was used for Gulf site evaluations. Computer modeling was used to predict increased noise levels during peak activity periods. In addition to construction and facility noise, noise and ground vibration from blasting was evaluated against thresholds of audibility and against community noise standards. A special study was conducted to assess wind and temperature gradient effects on noise propagation by D. Thomson of Pennsylvania State University for the Davis Canyon site. Major peer and technical reviews were performed on the acoustic modeling work.

3.4.3 Background Radioactivity

A background radiation survey was performed in Deaf Smith and Swisher Counties, Texas, in September 1982 (NUS, 1984a, BMI/ONWI-558). Existing data were used for Paradox and Gulf site evaluations. Pathway analyses and dose assessments were performed within performance assessment tasks.

3.4.4 Water Resources and Flooding

Extensive use was made of existing data on surface- and ground-water quality, precipitation, and current and projected water use of sites and locations in all three basins. Original borehole data, pumping test data, and water chemistry data from each site vicinity generated within geology and hydrology tasks were used to supplement the existing data base. Field surveys on stream (wash) channel cross sections were performed in Davis and Lavender Canyons in support of flooding studies which made use of HEC-1 and HEC-2 computer codes developed by the U.S. Army Corps of Engineers. Probable maximum and 500-year flood levels were predicted and plotted for the seven potentially acceptable sites. A special study was performed to access whether enough water to support repository construction and operation could be obtained for a Paradox site. Water rights and water allocations were reviewed with the Utah State engineers office. Water conservation and treatment schemes were conceptualized. Ground-water drawdown calculations were performed for the Richton and Deaf Smith sites to predict effects of project water pumping on nearby ground-water users.

3.4.5 Land Use

Project documents report extensively from existing land use data of the Gulf, Paradox, and Permian Basins. Data collection and analytic efforts focused on agricultural land use issues in the Permian Basin; grazing rights, State and national park, and cultural resource issues in the Paradox basin; and forest, agriculture, Camp Shelby Military Reservation, and town of Richton land use issues in the Gulf basin. Some verification in the field was performed for land uses of locations in all three basins. The project flew black-and-white and color infrared aerial photography of the Richton dome site in 1983. Black-and-white and color infrared aerial photography of the Deaf Smith and Swisher locations was flown in 1982 and 1985. Aerial photography and imagery from the French satellite, SPOT, was obtained for the Deaf Smith site and its vicinity (out to 30 kilometers for SPOT imagery) 1987. Historical aerial photography at numerous dates, from 1952 forward, were obtained. Full coverage for Deaf Smith and three other counties was obtained from the Agricultural Soil Conservation and Stabilization Service. Deaf Smith and Swisher County location data was digitized and managed within a computerized geographic information system (GIS). Land ownership and mineral rights were determined for all lands within the Davis Canyon, Richton Dome, and Deaf Smith sites.

3.4.6 Soils

Soil data were extracted from existing sources for project use. Designation of prime agricultural soils was determined formally with the U.S. Soil Conservation Service (SCS) using SCS Form AD1006 in compliance with the Farmland Protection Policy Act. The form process was completed for Richton Dome, Deaf Smith County, and Davis Canyon sites.

3.4.7 Aesthetics

Field surveys were made and photographs taken at the Richton Dome, Deaf Smith County, and Davis and Lavender Canyon sites to perform impact assessments on visual resources. Weather balloons were used to estimate visibility of proposed structures of the Richton Dome site. Five Paradox sites, and six alternative railway route options, were initially analyzed. Computer simulations and artist renderings were developed of repository facilities and six railway routes as viewed from scenic overlooks. The ASPEX computer program from Harvard University was used to determine which portions of structures and railway lines would be visible from selected viewpoints. About 1,500 square miles of Paradox basin terrain were digitized for computer analysis. Aesthetic resources for the Davis and Lavender Canyon sites were analyzed using the Bureau of Land Management Visual Resource Management (VRM) program (BLM Manual 8411) and the U.S. Forest Service VIEWIT computer codes. BLM performed the visual contrast rating analysis (BLM Manual 8431) for key observation points in and around Canyonlands National Park (BLM, 1985).

Visual resource characterization and impact assessment for the Deaf Smith site and vicinity were adopted from the BLM and U.S. Forest Service methodologies (NUS, 1984j, BMI/ONWI-561).

3.4.8 Cultural Resources

Extensive literature and file searches were conducted to characterize cultural, historic, and archaeologic resources on and in the vicinity of Gulf, Paradox, and Permian sites. These searches were supplemented with field surveys in the Paradox site vicinities and along seismic lines on Vacherie Dome. No recorded sites were identified at the Deaf Smith or Richton Dome sites. None of the 15 modern and historic cultural resources recorded during field surveys along seismic lines for Vacherie Dome were believed to be significant. Onsite surveys started at Gibson Dome in 1981 and ended with two major surveys in the Davis and Lavender Canyon areas: one survey consisted of a BLM Class III (intensive) survey of a 640-acre plot in Davis Canyon; the other was a BLM Class II (10 percent sample) for a 20,800-acre sample area. Forty-five sites were discovered or assessed in the Davis and Lavender Canyon project area, and 11 of these are believed to be eligible for nomination to the National Register of Historic Places. Of particular interest was a fire hearth dating from 900 B.C., rock art sites, and a group of structures with potential astroarchaeological significance.

3.4.9 Aquatic and Terrestrial Ecology

Field identification and evaluation of potentially important ecological resources that could constrain repository siting were performed in the vicinities of Paradox, Gulf, and Permian sites. Plant community and threatened and endangered species field surveys were made in the Gibson Dome area in 1982 and 1985 and in the Elk Ridge study area in 1982. Threatened and endangered species field consultations were made in the vicinity of Mississippi and Louisiana sites in 1983. A survey of T&E species at proposed field activity sites in Cottonwood Creek Canyon and Indian Creek Canyon, Utah, were made in 1983. Davis Canyon transportation corridors were also field surveyed (inspected) in 1983 and a field reconnaissance was conducted for the Davis Canyon site and proposed road and rail access routes in 1985. Wetland determinations were made for Deaf Smith County, Richton, and Davis Canyon sites. A T&E reconnaissance survey was also performed for planned seismic lines over Vacherie Dome. Recognized local experts were used in all of these studies.

3.4.10 Salt Studies

Field studies to better understand the behavior of salt in the environment included visits to the WIPP salt pile and a salt particle size distribution field study at the International Salt Company's Retsolf Facility, near Genesee, New York. Results from the 1984 two-week program that measured wind-carried salt particles proved inconclusive and indicated development was needed on the techniques and equipment used to measure and observe salt. Several computerized literature searches were made to identify studies of salt impacts on vegetation and to characterize salt deposition from facility operations. A scoping study and special study to assess options for salt management and disposal were performed (NUS, 1986a,b). These studies were Permian Basin evaluations and were applicable, in part, to the Gulf and Paradox sites.

3.4.11 Transportation

Field and literature data from many of the above disciplines were used to identify rail and road access routes to the Paradox and Permian sites. Access to Gulf sites was straightforward and not studied extensively. Field surveys of track and road conditions were made in the vicinity of all seven potentially acceptable sites. An intensive GIS-based alternative rail access route identification process was conducted for the Deaf Smith County site (NUS, 1986c). For Paradox sites, extensive field reconnaissance, environmental, and conceptual engineering studies were performed to establish the feasibility and cost of rail and road access. Field reconnaissance via helicopter and 4-wheel drive vehicles was made of alternative routes before they were finalized. Plans and profiles for 180 miles of railroad and 25 miles of highway, cut-and-fill volumes, bridge types, tunnel requirements, excavation methods, and construction costs were estimated or prepared. Ultimately, more than 1,100 route miles of potential corridor alignments were studied and costed in demonstrating feasibility, identifying alternative routes, and in assessing impacts of rail and road access for the Paradox sites (BGI, 1982b).

3.4.12 Socioeconomics

All socioeconomic program accomplishments, except EA inputs and the Socioeconomic Monitoring and Mitigation Plan, are discussed in this section, because the work took place in WBS 3.0. Socioeconomic data were collected from existing published sources and by interviewing local community people. Information was collected on the topics of education, health care, human services, water and sewer facilities, solid waste, government revenues and expenditures, labor force, migrant workers, and city growth plans. Socioeconomic data base reports were developed for multi-county areas in Utah, Louisiana, Mississippi, and Texas and published in ONWI reports. About three million dollars was spent developing, peer reviewing, and refining the Socioeconomic Analysis of Repository Siting (SEARS) model. Social assessment methods were recommended for use in the program (ONWI, 1986c).

The SEARS model was developed with the support of the Department of Rural Sociology, Texas Agricultural Experiment Station (TAES, 1984a-e; 1985a,b) at College Station, Texas, and the Department of Agricultural Economics at North Dakota State University. Seven major reports on the SEARS model and its use were prepared. Economic, demographic, economic-demographic interface, social service, and fiscal modules were built into SEARS. An external peer review panel, comprised of experts of national prominence and chaired by Dr. Christopher Cluett, reviewed the SEARS model and its documentation (Cluett, 1985).

A similar panel chaired by Richard P. Gale reviewed the Social Assessment Methods Recommendation Report (Gale, 1986). A "grants equal to tax" model (GETT), later converted to "payments equal to tax" (PETT) model, was adapted for program use (ONWI, 1986a,b). In 1981, a Framework for Community Planning (ONWI, 1981) was developed as a tool to facilitate discussions with communities potentially affected by the site characterization studies and the repository. An update planned for 1987 remains incomplete.

Impact analyses addressed effects of population in-migration on repository site areas, potential effects on water use, and impacts on agribusiness in Deaf Smith County and vicinity. Payment-equal-to-tax projections were made. Technical workshops for State and local groups on impact analysis methods, and community planning meetings with State and local officials on mitigation strategies, were held. Survey instruments to monitor socioeconomic impacts were developed and tested. The above accomplishments were organized under impact assessment, mitigation, and monitoring categories.

3.4.13 Impact Assessment

The studies discussed above provided data and analyses to support the NWPA (1982) EAs. Analyses performed are represented by the range of topics listed in the "contribution" column of Table 3-4.

3.5 PLANS TO IMPLEMENT SITE CHARACTERIZATION ACTIVITIES

This section presents the status of plans that were to be implemented to acquire field and laboratory data on the geotechnical, environmental, and socioeconomic characteristics of the salt site in Deaf Smith County, Texas.

3.5.1 Basis for Plans

The scope and purpose of the plans are based on a hierarchy of documents, including the following:

1. SRPO (Salt Repository Project Office), 1987. Salt Repository Project Requirements Document, SRP/B-18.

2. DOE (U.S. Department of Energy), 1986. Issues Hierarchy for a Mined Geological Disposal System, OGR/B-10.

3. DOE (U.S. Department of Energy), 1988. Site Characterization Plan, Deaf Smith County Site, Texas, Consultation Draft, Volume 4, DOE/RW-0162.

4. ONWI (Office of Nuclear Waste Isolation), 1987. Salt Repository Project Surface Investigation Plan, May 1987, Revision 1, Draft, Battelle Memorial Institute, Columbus, OH.

Collectively, these documents include the scope of actions needed to comply with regulations and statutes and for acquiring the types of data to be used for resolving design and performance issues.

3.5.2 Status of Plans

The U.S. Department of Energy required that all technical field programs be performed on the basis of documented implementation plans. In accordance with that requirement, a series of Site Study Plans (SSPs) and Environmental Field Activity Plans (EFAPs) were prepared. The plans were to be implemented

Table 3-4. Technical Analyses and Input for Final EAs for Deaf Smith County, Davis Canyon, and Richton Dome Sites (Page 1 of 2)

Discipline	Section	Contribution
Repository Site Selection	2.4	Site comparison methodology and application
Land Use	3.4.1	Land-use patterns; land ownership
	4.2.1.1	Effects on land use; individual land owners
	5.2.3	Effects on agricultural land; irrigation; prime agricultural soils; salt deposition
Terrestrial and Aquatic Ecosystems	3.4.2	Potential natural habitat; T&E species; sensitive areas; ecosystems; critical/unique habitats
	4.2.1.2	Effects on local ecological resources
	5.2.4	Evaluation of aquatic and terrestrial ecosystems and T&E species
Air Quality/ Meteorology	3.4.3	Air quality; climatological summaries; severe weather; atmospheric transport diffusion
	4.2.1.3	Analysis of air quality: modeling with ISC
	5.2.5	Emissions and effects on air quality
Noise	3.4.4	Background sound levels
	4.2.1.6	Increase of sound levels modeling
	5.2.7	Modeling of noise impacts
Aesthetic Resources	3.4.5	Visual variety classes
	4.2.1.7	Visual character of the landscape
	5.2.6	Visual character of the landscape: Visual/Contrast ratings
Cultural and Historical Resources	3.4.6	Archeological, cultural, and historical resources
	4.2.1.8	Implementation of PMOA
	5.2.8	Indirect and secondary impacts
Soils	3.2.9	Soil associations
	4.2.1.5	Potential impacts and reclamation
	5.2.1.1	Potential impacts and reclamation
Transportation/ Utilities	3.5	Pattern and traffic capacity; airports; waterways; utilities
	4.2.1.10	Expected increase in traffic
	5.3.2	Effects of improvement to transportation corridors
	5.3.3	Effects on transportation infrastructure
	5.3.4	Utility requirements and impacts

Table 3-4. Technical Analysis and Input for Final EAs for Deaf Smith County, Davis Canyon, and Richton Dome Sites
(Page 2 of 2)

Discipline	Section	Contribution
Hydrology	3.3.1	Surface-water resources and quality
	3.3.3	Water supply; current and projected water use
	4.2.1.4	Water consumption, supply and quality; flooding analyses
	5.2.2.1	Surface water supply and quality
Salt Management and Disposal	4.2.1.11	Impacts and measures for mitigation
	5.2.10	Potential impacts; disposal options
Background Radiation	3.4.7	Background radiation levels
	5.2.9	Radiological impacts
Socioeconomics	3.6	Population density, distribution, projections; economic conditions: employment, income, agriculture; community services: housing, education, health, recreation, protective water, sewer, solid waste; social conditions: history, lifestyle, attitudes, social problems, government and fiscal conditions
	4.2.2	Modeling of effects of site characterization
	5.4	and repository development on population, economy, community services, social conditions, and government and fiscal conditions using SEARS and Grants Equal to Taxes Models
	4.4.1	Cumulative site characterization impacts on Canyonlands National Park (Davis Canyon only)
	5.5.1	Cumulative repository impacts on Canyonlands National Park (Davis Canyon only)
Site Suitability Analysis	6.2	Evaluation of suitability of sites related to population density and distribution, meteorology, offsite installations and operations, environmental quality, socioeconomic impacts, and transportation guidelines from 10 CFR 960.

Chapter 2 sections dealt with site selection.
Chapter 3 sections dealt with the site and vicinity characteristics.
Chapter 4 sections dealt with impacts of site characterization.
Chapter 5 sections dealt with impacts of a repository.
Chapter 6 sections evaluated site characteristics against the 10 CFR 960 DOE siting guidelines.

for characterizing the Deaf Smith County, Texas, salt site. The titles and status (December 1987) of these plans are given on Table 3-5. An SSP is a field implementation plan which describes studies designed to collect data to satisfy information needs contained in the Site Characterization Plan (SCP). An EFAP is a field implementation plan which describes activities designed to collect data to satisfy information needs contained in the Environmental Monitoring and Mitigation Plan (EMMP) or Environmental Regulatory Compliance Plan (ERCP). Technical Procedures (TPs) involved in the conduct of planned work are listed in their respective SSP or EFAP. TPs were not prepared for socioeconomic plans.

The contents of the plans listed in Table 3-5 include the field, office, and laboratory activities and methods, the rationale for the activities and methods, and relating data needs to test methods. The data needs are responsive to the issues in the Site Characterization Plan and the Surface Investigation Plan (Items 3 and 4 above).

Table 3-5. Titles and Status of Geotechnical and Environmental Plans
(Page 1 of 2)

Title	Status
Engineering Design Boreholes No. 1 and No. 2 SSP[a]	Reviewed Preliminary Draft, November 1987
Engineering Design Boreholes Seismic Surveys SSP	Reviewed Preliminary Draft, July 1987
Upper Aquifer Hydrology Clusters SSP	Working Draft, December 1987
Intermediate Hydrology Clusters SSP	Reviewed Preliminary Draft, November 1987
Lower Aquifer Hydrology Clusters SSP	Working Draft, December 1987
Exploratory Shaft Facilities Design Foundation Boreholes SSP	Reviewed Preliminary Draft, July 1987
Laboratory Soil Mechanics SSP	Reviewed Preliminary Draft, September 1987
Routine Laboratory Rock Mechanics SSP	Reviewed Preliminary Draft, September 1987
Non-Routine Laboratory Rock Mechanics SSP	Reviewed Preliminary Draft, September 1987
Exploratory Shaft Monitoring Wells SSP	Reviewed Preliminary Draft, July 1987
Borehole Search and Characterization SSP	Reviewed Preliminary Draft, July 1987
Geochemical Sampling Requirements and Methodologies Plan	Working Draft, December 1987
Geochemical Analytical Requirements and Methodologies Plan	Working Draft, December 1987
Socioeconomic EFAP[b]	Draft responded to SRPO comments, July 1987
Water Resources EFAP	Draft responded to DOE-HQ and SRPO comments, December 1987

Table 3-5. Titles and Status of Geotechnical and Environmental Plans
(Page 2 of 2)

Title	Status
Water Resources SSP	Draft responded to DOE-HQ and SRPO comments, December 1987
Meteorology/Air Quality EFAP	Draft responded to DOE-HQ and SRPO comments, December 1987
Meteorology/Air Quality SSP	Draft responded to DOE-HQ and SRPO comments, December 1987
Land Use EFAP	Draft responded to SRPO comments, December 1987
Land Use SSP	Draft responded to SRPO comments, December 1987
Background Environmental Radioactivity SSP	Draft responded to ONWI and SRPO review, July 1987
Acoustics EFAP	Draft responded to DOE-HQ and SRPO comments, December 1987
Cultural Resources EFAP	Draft responded to DOE-HQ and SRPO comments, December 1987
Ecology EFAP	Draft responded to DOE-HQ and SRPO comments, December 1987
Salt EFAP	Draft responded to ONWI comments, September 1987
Soils EFAP	Draft responded to SRPO comments, December 1987
Aesthetics SSP	Was to be made an EFAP. Draft responded to ONWI and SRPO comments, July 1987
Utilities/Solid Waste SSP	Was to be made an EFAP. Draft responded to ONWI and SRPO comments, July 1987
Transportation SSP	Was to be made an EFAP. Draft responded to ONWI and SRPO comments, July 1987

a. SSP = Site Study Plan.
b. EFAP = Environmental Field Activity Plan.

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CHAPTER 4

REPOSITORY (WBS 4.0)

The purpose of this chapter is to provide an overview of the activities performed as part of the Salt Repository work element. This overview consists of a scope description, results achieved to date, and work that was planned at the time of project termination for each repository element.

The Repository work element consists of those technical activities required to define and develop the repository system components (i.e., surface and subsurface facilities, shafts, equipment, seals, structures, construction, operations, decommissioning, costs, and schedule estimates) and to develop the tools and methodologies required to analyze, test, and monitor the performance of the repository (ONWI, 1987a).

These technical activities have been organized into the following categories:

1. Rock Mechanics. Laboratory tests were developed and conducted to determine time-dependent properties of the host rock required to support repository design and performance assessment. Preliminary testing methods and modeling tools were developed. Using models and test data, thermomechanical analyses of the repository host rock were performed to support salt site environmental assessments and the Deaf Smith County site characterization plan (SCP).

   The assessment of the results of the laboratory rock mechanics tests is that the most significant parameter influencing the repository design is the potential for creep of the host rock under anticipated repository temperatures and stresses. The salt project intended to resolve the impact of this phenomenon on the repository design and performance assessment during the site characterization phase.

   The salt project developed a preliminary model of host rock behavior under anticipated repository conditions. Further model development was planned to incorporate site-specific data in order to achieve a validated model.

2. Equipment and Instrumentation. Equipment and instrumentation needs for the following repository activities were evaluated: waste handling, processing, emplacement and retrieval, subsurface excavation, postclosure seal and backfill emplacement, and monitoring of repository construction and operations. Based on the SCP conceptual design, the following equipment/instrumentation items were identified as requiring further development during subsequent design phases:

   - Cask inspection manipulator

   - Remote decontamination components

- Cask opening manipulator
- Spent fuel disassembly machine
- Fuel assembly hardware compaction machine
- Canister/waste package container welder
- Canister/waste package container weld inspection manipulator
- Canister/waste package container contamination inspection/decontamination components
- Special process equipment (waste handling building)
- Surface/underground cage on/off loader
- Underground transfer cart/transporter
- Waste package emplacement machine
- Shield plug emplacement machine
- Emplacement room backfill machine
- Waste package location instrumentation (retrieval)
- Waste package excavation/extraction equipment (retrieval)
- Postclosure monitoring instrumentation.

3. Sealing. Candidate materials for repository long-term seals have been tested under laboratory conditions. Results from these tests were used to support repository design and performance assessment. Preliminary test methods for further testing of seal materials, as well as seal performance models, were developed. Sealing conceptual designs were incorporated into the salt environmental assessments and the repository SCP conceptual design.

   The results of the seal system performance analyses indicate no adverse consequences are anticipated. Further model development and testing (both laboratory and in situ) are needed to confirm these preliminary assessments.

4. Facility Design. Over the course of the salt project, several design concepts were developed, most of which were generic in nature owing to the site selection focus of the project. With the selection of the Deaf Smith County site in 1986 (based on the environmental assessment of several salt locations) as the preferred salt site, a site-specific conceptual design was developed. This design was intended to support the NWPA requirement that site characterization activities be based on a repository concept.

The conceptual design features a 4,000-acre underground area (supported by a 600-acre surface facility) to dispose up to 70,000 MTU of high-level waste/spent reactor fuel over a 25-year period. Access to the subsurface would be provided by a series of lined shafts. After emplacement of all the waste, the repository would be sealed and isolated from the accessible environment.

Future design phases were to be performed to support preparation of a license application (for approval to start construction) and, subsequently, an application for an operating license.

The repository design phases were intended to be integrated into other project activities, such as the license application, site characterization testing, performance assessment, modeling, and equipment development. If the salt site was chosen for the high-level waste repository, the final design phase was intended to be the basis for repository construction and operation.

5. Performance Assessment. The performance of the repository system in terms of meeting safety and isolation requirements has been simulated in a preliminary performance assessment model. This model represents the ability of the repository to isolate wastes from the accessible environment under anticipated conditions over long time periods. The potential releases during operation and postclosure phases have been evaluated in terms of potential radiation exposures to the public.

   The performance assessment model is based on the Deaf Smith County site environmental assessment, site characterization plan conceptual designs, rock mechanics, waste package, and sealing models, as well as available near-site geologic data. The results of performance assessments of the repository were to have been integrated into subsequent phases of project development (i.e., design, characterization testing, licensing, model development, etc.).

   Results to date indicate that the salt repository would meet the EPA (EPA, 1982) and NRC (NRC, 1980) requirements by considerable margins.

   The following sections of this chapter describe the status of each of the repository work elements in greater detail.

4.1 ROCK MECHANICS

This subelement includes the laboratory measurement and characterization of those properties of the salt rock units at the repository horizon that will change with time following subsurface development, waste emplacement, or decommissioning. In addition, this element of work will develop, benchmark, and validate the models predicting this time-dependent behavior. (Note: Rock Mechanics laboratory measurement of salt rock index properties, time-independent salt rock unit properties, and properties of non-salt rock units are included in Chapter 3, Site (WBS 3.0).)

The scope includes very-near-field (waste package) and near-field (repository room scale) effects on mine stability as well as far-field (repository site scale) effects relative to the overall integrity of the geologic containment of high-level waste (Oschman et al., 1987). The rock mechanics models themselves are used in both the design of subsurface openings and in the licensing of that design. A total of three models were planned for development. In the laboratory testing to support the development of these models, the effects of radiation on salt materials properties and material response were to be evaluated and their effects included in the models.

At present no site-specific data were available except for the topographic studies performed within the 23.3 km^2 site. Hence, all the available geoengineering data were obtained from offsite sources such as test data on rocks and salt samples obtained from program wells representative of the Palo Duro Basin (basin-wide data); wells representative of Deaf Smith County; and Avery Island Mine, New Iberia, Louisiana (Avery) domal salt.

Within the program, in situ tests have been performed in the offsite test sites such as at Avery Island, Asse Mine at Asse in the Federal Republic of Germany, and the data published by the Waste Isolation Pilot Plant (WIPP) near Carlsbad, New Mexico. At these sites, geomechanical, thermomechanical, geochemical, and geohydrologic tests have been conducted.

4.1.1 Accomplishments

4.1.1.1 Laboratory/Field Test Results

The mechanical properties of intact rock identified from the program wells have been categorized into six rock groups: salts, chaotic salts, anhydrites, salty anhydrites, clastics, and carbonates. Strength tests indicate that the carbonates and anhydrites are strong to very strong, whereas the clastics vary from very weak to strong. (Clastics generally become more competent below the Dockum.) The Palo Duro Basin salts have relatively higher ductility than other salts.

Few discontinuities were observed in the drill core and there is little or no evidence for discontinuities in the host rock salt, other than bedding plane joints and interbeds. Too few shear tests have been performed to enable meaningful conclusions to be drawn.

Apart from some geophysical data, no site data are available relevant to large-scale mechanical properties, because no large-scale testing has been performed in the vicinity of the site. Because of a scarcity of data and uncertainties in the large-scale data obtained for nonsalt rocks at other sites, no projections can be made concerning the large-scale behavior of nonsalt rocks at the Deaf Smith County site.

The average thermal conductivity of Palo Duro salts tested under pressure varies from approximately 5.40 W/(m·K) at 30°C to 3.56 W/(m·K) at 200°C, a decrease of about 35 percent. Thermal diffusivity is independent of pressure in the range 0 to 30 megapascals (MPa) and varies from approximately

2.64×10^{-6} m^2/s at 30°C to 1.71×10^{-6} m^2/s at 200°C, a decrease of about 35 percent. The specific heat of Lower San Andres salts is approximately 900 J(kg K). Temperature-dependent values of the coefficient of thermal expansion have not been determined, but the average value of Palo Duro salt is approximately 35×10^{-6}/K for the temperature range 25 to 200°C for specimens confined at pressures between 3 and 25 MPa. Over the range of confining pressure 3 to 25 MPa, pressure has no detectable influence on the coefficient of thermal expansion.

Except for salt, rocks at the site are not expected to exhibit significant temperature dependence over the small range of temperatures anticipated in the repository region.

Creep and other similar time-dependent aspects of salt behavior, such as stress relaxation, are significant over the ranges of stress and temperature anticipated in the repository. Salt hardens as it deforms, and time-dependent behavior dominates the material response of salt. Under repository conditions, the concept of a yield stress generally does not apply. In the long term, salt behavior is influenced by stress difference and temperature.

Tests carried out at various salt mines reveal the large-scale behavior of salt in general, but do not yield information that can be directly applied to the Deaf Smith County site.

Within the nonsalt rocks of the Palo Duro Basin, the major principal stress direction is vertical; the other principal stresses are horizontal and nearly equal. In salt, it appears that the stress regime is approximately uniform (or hydrostatic) and equal to the overburden pressure.

The behavior of crushed salt, thermal decrepitation, thermal dewatering, fracture healing, the effects of interbeds, and the effects of moisture on mechanical behavior are not yet fully understood. Although most of the special geoengineering properties can be expected to influence repository design and performance assessment, much more work is required to characterize the processes involved. This work was to be the subject of future studies.

The stress to which the repository rocks will be subjected depends upon the existing stress regime, the stress changes due to excavation of the repository, and thermomechanical stress changes due to waste emplacement. An understanding of all these factors, together with accurate knowledge of the applicable geoengineering properties, is necessary for the design, safe operation, and long-term performance of the underground repository.

4.1.1.2 Rock Mechanics Models

The constitutive model (or material model) should represent stress-strain behavior (derived from the test results) in a mathematical relationship that is applicable to the ranges of stress and temperature expected for the nuclear waste repository. It should take into account at least the three main components of salt deformation, i.e., elastic deformation, thermal expansion, and inelastic deformation. Although no material model for salt is universally accepted at the present time, the Salt Repository Project Office (SRPO) has selected the exponential-time model for use in design and performance assessment calculations. This is a semi-empirical model based on first-order kinetics. Two limitations of the model are that it does not account for brittle deformations due to microcracking, and that it oversimplifies the effects of stress and temperature on viscoplastic deformation.

Present Status of Material Modeling Efforts (December 1987): The exponential-time model has several limitations. First, the exponential-time model does not account for brittle deformation resulting from microcracking. Also, the functional form for modeling stress and temperature dependence of viscoplastic deformation has been found to be too simple to reproduce accurately the behavior of salt over the ranges of stress and temperature expected for a nuclear waste repository. For example, this model was unable to reproduce the behavior of corejack tests performed at the Avery Island mine in Louisiana. A brief discussion of the limitations of the exponential-time model may also be found in Oschman et al. (1987, pp. 34-35). Other material models (i.e., Munson-Dawson and Krieg models) were being studied as potential candidates to replace the exponential-time model.

In order to accommodate the limitations of the exponential-time model certain strategies were adopted. Sensitivity analyses of the parameter values of the model were performed so that a simulated conservative case could be adopted during the various design-modeling activities. Modeling results obtained from the exponential-time model were compared with actual measurements obtained from underground operations to determine qualitatively just how well the model performed. In general, however, since most of the performance analyses and design analyses performed to date were preliminary in nature, the impact of the inaccuracies of the exponential-time model was not great. Greater accuracy, however, was becoming more important as site characterization neared. Site characterization activities should have provided much of the information necessary for an accurate formulation of the new models, thereby resolving the problems of utilizing the exponential-time model.

The two material models mentioned previously were proposed as interim replacements for the exponential-time model. These models, the Munson-Dawson and the Krieg models, were selected after a careful survey of the literature on salt and other polycrystalline materials, and an extensive model-fitting activity. This model fitting and selection was accomplished on existing Avery Island salt because of its purity, homogeneity, and the size of the testing data base already in existence. Once this less complex salt had been modeled, it was anticipated that the next step to site-specific LSA-4 salt could be accomplished more effectively and quickly. The model evaluation and modification on Avery Island salt was expected to continue during the next few years as data from ongoing tests were included in the modeling effort.

The Krieg and Munson-Dawson models are presented here in their present stage of development. They were both in the process of evaluation and modification and would have continued to be so for the next several years, even after they eventually became baselined.

The Munson-Dawson constitutive model is based on the deformation mechanisms that are believed to control steady-state deformation over the ranges of stress and temperature of interest for a nuclear waste repository (Munson and Dawson, 1982). The micromechanisms that are incorporated into this model are dislocation glide, dislocation climb, and an undefined mechanism.

Transient deformation is modeled empirically using the concept that the approach to steady state will be different depending on whether the material is hardening or recovering.

The Krieg constitutive model (Krieg, 1982) uses the internal variable α_{ij} to incorporate thermomechanical history. These variables have the dimensions of stress and are referred to as the backstress. Micromechanically, the backstress can be related to the mobile dislocation density. Hardening (or softening), such as occurs during transient creep, results from an increase (or decrease) in the mobile dislocation density. Steady state is reached when the hardening and softening balance and the backstress becomes constant.

As both models evolved further, successive versions would be baselined. As more and better data became available, these models quite probably would be changed.

There was, in addition, a separate material modeling activity. Rather than selecting an existing model, a model was being developed from first principles. It represented a different philosophy, which would ensure that no approach to formulating a viable material model for salt was overlooked. It was in a much earlier phase of development than either the Krieg or the Munson-Dawson model and would have been presented when its preliminary form was more complete.

A general constitutive (material) model for salt should include the contribution of four deformation components, namely, the elastic, thermal, viscoplastic, and brittle components. The models just presented describe only the viscoplastic component of the deformation. The elastic and thermal components are relatively well understood, and their incorporation into an overall constitutive model is reasonably straightforward. The development of a model to describe the brittle component is far less advanced than that for viscoplastic deformation. It was anticipated that the final outcome of the overall material modeling program would be a model that includes all four of these components.

Parametric studies of mechanical properties were performed for several nonsalt horizons in the Detten and Zeeck wells in the Palo Duro Basin. These tests were performed to determine elastic properties, strength and strength parameters, friction angle and cohesion. The samples were chosen from relatively homogeneous horizons that may not be representative of the rock unit as a whole.

During the course of the site investigation studies, rock property values were determined for over 1,600 test specimens. A statistical summary was deemed appropriate to approach this data.

4.1.2 Planned Work

The performance of the overall rock mechanics program is dependent on integrating the following activities:

- Laboratory testing (including the supporting analysis and numerical modeling)
- In situ testing and analysis (surface and subsurface)
- Design and performance assessment.

The rock mechanics testing required must satisfy Federal regulations, design data needs, and performance assessment data needs.

Plans included the laboratory characterization of index properties, and hydrologic, mechanical, and thermal behavior of all rock units that may be encountered above and below the repository. The information gathered from laboratory studies was to be used in the preparation of State permit applications, design calculations, and performance assessment analyses. Federal regulations also require that specific technical concerns and issues be addressed before a license for repository operation is granted. Specific procedures and detailed matrices for the planned tests are outlined in Oschman et al. (1987).

The activities associated with the development of a more comprehensive model for salt included a literature survey to provide a number of candidate mathematical forms for the material model. These candidate forms were to be screened based on their ability to fit the existing Avery Island salt data base (Avery Island salt is the logical choice because it is the purest natural salt available, thereby introducing the fewest complexities that might interfere with an understanding of the mechanical behavior of salt). The models that survive the screening were to be used to predict laboratory, bench-scale, and field tests that have already been or are still to be performed on Avery Island salt. Models that survive this second screening would be revised to include information derived from new laboratory testing. These models would then be rescreened. Models that survive this final screening would be used for evaluation of salt from a candidate repository site.

A laboratory testing program was to be conducted to determine the time-dependent thermally coupled mechanical behavior of salt (some of this testing had already been accomplished as of December 1987). This salt testing program had several purposes:

- Evaluation of a new salt material model
- Assessment of anisotropy on the mechanical properties
- Assessment of the influence of stress history on deformation
- Determination of the influence of impurities on the mechanical behavior.

This testing program included several types of mechanical testing, the principal types being:

- Creep tests (constant stress)
- Relaxation tests (constant strain)
- Constant stress-rate tests (quasi-static)
- Constant strain-rate tests.

It was anticipated that the above-mentioned tests would be performed under uniaxial and triaxial compression conditions and some possibly in tension. The results of these tests will provide information to determine the load path dependency of the material model. Various loads, strain-rates, and temperatures would be investigated in order to explore the range of anticipated repository conditions. Both increments and decrements in stress and strain would be introduced into the tests and any model presented would have to account for all of the above load path behaviors. Additional information derived from micro-mechanical studies would be incorporated into the modeling effort.

Figure 4-1 shows activities that are involved in the conceptual modeling process. Initially, parameters essential to repository design, performance assessment, and resolution of issues are identified. Conceptual models are then developed to quantify these parameters. In some instances, it may be necessary to develop and implement a special conceptual model to quantify a specific parameter; however, it is generally possible to develop and implement a conceptual model to determine a group of related parameters. Information and data needed as input to the conceptual models are subsequently identified and gathered mainly from laboratory and field testing programs during repository site characterization. Calculations using the conceptual model can then be performed. These can then be validated by comparing them against field results. Final calculations can be made, and the results used for design, performance assessment, and the resolution of issues.

If the conceptual model requires development of computer codes or the use of existing codes, these need to be verified as well as validated. As these codes are modified, they will need to be reverified and revalidated.

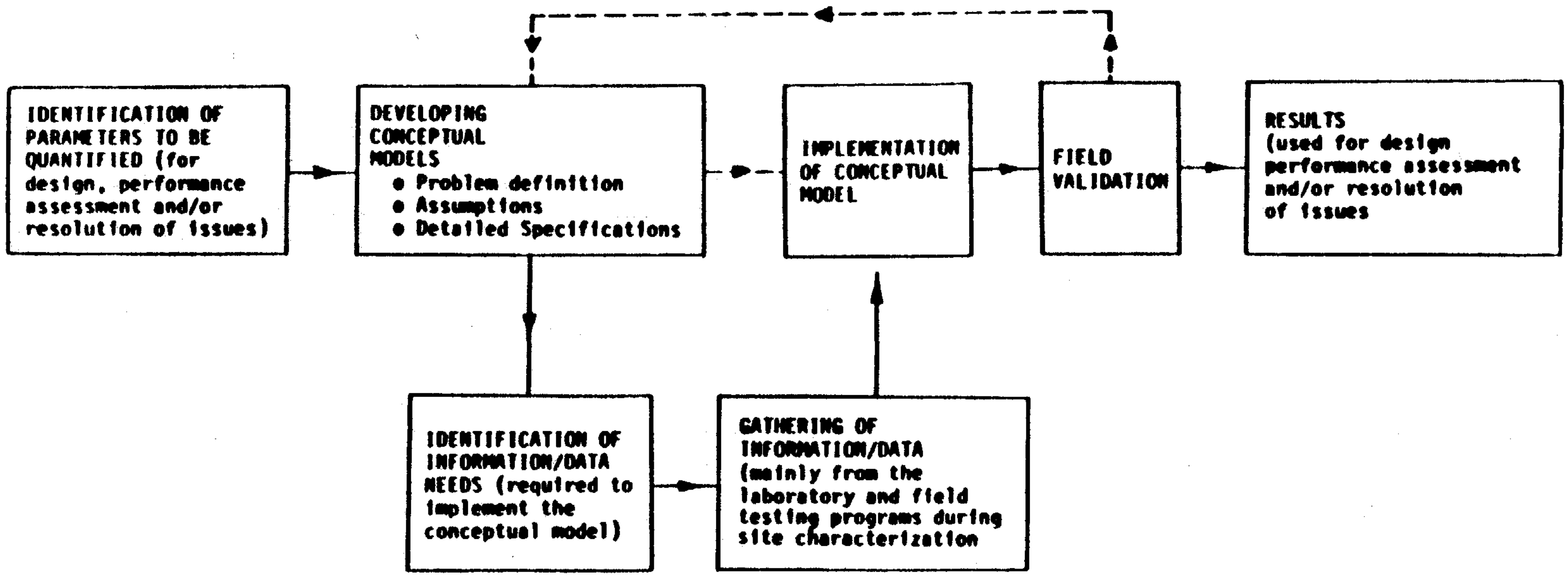
4.2 EQUIPMENT AND INSTRUMENTATION

This subelement of work includes (ONWI, 1987a):

- The analysis of repository equipment and instruments to identify those items that are not based upon demonstrated technology or demonstrated operation

- The design, development, fabrication, and testing of equipment and instrument prototypes for inclusion in repository designs, thus providing the technology demonstration required by regulations.

Both Department of Energy (DOE, 1985b) and Nuclear Regulatory Commission (NRC, 1980) regulations require repository equipment designs to be based upon demonstrated technology. This WBS subelement will develop equipment performance requirements for all phases of repository operations, including



Activity Chart Showing the Role of Conceptual Models in Design, Performance Assessment, and Resolution of Issues

Figure 4-1

waste receiving, waste handling, waste packaging, waste emplacement, subsurface development and mining, decommissioning seal emplacement, and waste retrieval. The results will be used in the repository design and in equipment procurement bid packages.

4.2.1 Accomplishments

Laboratory testing of prototype stress and displacement measurement sensors was completed. Sensors of this type were planned to be used in the in situ test program.

A draft repository SCP conceptual design was completed (SCP-CDR, Fluor, 1987c,d). Included in this design are concepts for repository systems and equipment that receive, handle, package, and emplace waste; the systems and equipment that retrieve waste; and the systems and equipment used to develop subsurface workings.

Based on the SCP-CDR (Fluor, 1987c,d), the following equipment/instrumentation items were identified as requiring further development during subsequent design phases:

- cask inspection manipulator
- remote decontamination components
- cask opening manipulator
- spent fuel disassembly machine
- fuel assembly hardware compaction machine
- canister/waste package container welder
- canister/waste package container weld inspection manipulator
- canister/waste package container contamination inspection/decontamination components
- special process equipment (waste handling building)
- surface/underground cage on/off loader
- underground transfer cart/transporter
- waste package emplacement machine
- shield plug emplacement machine
- emplacement room backfill machine
- waste package location instrumentation (retrieval)
- waste package excavation/extraction equipment (retrieval)
- postclosure monitoring instrumentation.

A draft salt project retrieval strategy paper was prepared (Tomé et al., 1987). Included in this document are plans to develop equipment to accomplish retrieval of waste. This development includes proof-of-principle and prototype demonstration testing. In general, these activities support repository design by defining repository equipment requirements and the prototype testing of that equipment.

4.2.2 Planned Work

A major objective of the repository advanced conceptual design was to be the schematic design and definition of all repository systems, facilities, and equipment. These schematic design activities include the following:

- Conceptual performance requirements and design definitions for waste-processing, handling, and emplacement systems and equipment were to be developed. Results were to be documented in conceptual mechanical flow diagrams, process descriptions, and performance data sheets.

- Conceptual performance requirements and design definitions for the subsurface excavation systems and equipment were to be developed. Results were to be documented in conceptual mechanical flow diagrams, process descriptions, mine development plans, and performance data sheets for major subsurface excavation systems and equipment.

- Conceptual designs for repository monitoring and control systems were to be developed. These systems include geotechnical, radiation, environmental, and process monitoring and control systems.

- Conceptual performance requirements and design definitions for repository waste retrieval systems were to be developed. Results were to be documented in conceptual mechanical flow diagrams, process descriptions, and retrieval equipment specifications.

Prototype testing of repository equipment was to be conducted under this subelement. The scope of this testing work was to include the design, fabrication, and testing of repository equipment prototypes. To date, limited prototype equipment development and testing has occurred in the salt program. Equipment requiring prototype testing was to be identified during the repository advanced conceptual design phase. Prototype equipment testing would also involve seal emplacement equipment.

Spent fuel consolidation and waste form canisterization have the highest risk of radiological emissions (ONWI, 1987d). The DOE is sponsoring a program at the Idaho National Engineering Laboratory to develop spent fuel consolidation systems and equipment designs and to test those designs. Data will be obtained from these demonstration programs that define routine and accident emissions. Prototype equipment design, fabrication, and testing will also be completed for other waste-handling, packaging, and emplacement equipment. Radiological emissions data will be obtained as part of this prototype testing program as well.

Engineering and design trade-off studies were to contain capital and operating cost data. Equipment and instrumentation activities were to support preparation of these cost estimates. Prototype equipment test reports were to contain development equipment capital and operating cost data. The resolution of licensing issues will result from the selection of cost-effective alternative repository designs that meet appropriate regulations and DOE requirements. Generally, these engineering trade-off studies and alternative concept selections will occur during advanced conceptual design. However, some additional trade-off studies may be required during license application design because of new knowledge gained during site characterization.

4.3 SEALING

The disposal of high-level nuclear waste in a deep geologic repository will require penetrating the geologic setting with boreholes, shafts, drifts, and storage rooms that may be potential pathways to the accessible environment. These penetrations, which may already exist from previous exploration activities, must be sealed adequately by a seal system. This seal system will prevent ground-water flow into the repository and radionuclide release into the accessible environment in quantities in excess of acceptable levels.

This subelement involves three main types of activities:

- Research, development, and testing. R&D efforts focus on the identification and selection of seal materials and the development of a data base for the selected seal materials. R&D efforts to date have selected earthen (clay), cementitious, and salt materials for the engineered barriers.

- Seal design. Designs were to be developed for all repository decommissioning seal systems. Design activities for each seal system will include seal materials selection, seal material specifications, seal emplacement design, and seal emplacement equipment specifications.

- Seal system performance analysis. The key performance variable in seal design is water migration through the engineered seal barriers. Models that predict fluid flow rates through each seal system design were to be developed and benchmarked.

In all cases, sealing must address three seal-zone components, each of which is a potential leakage path. These components are: (1) seal materials, (2) interfaces between seal materials and host rock and between seal materials and liners, and (3) disturbed zones in the host rock.

The seal system was to be constructed using two basic components: bulkheads and backfill. These two components were to be fabricated from run-of-the-mine crushed salt; cementitious grout and concretes; or earthen materials, including crushed rock, gravel, sand, and clay. More specifically, four components were to be used in the seal system: (1) bulkheads constructed with cementitious materials, (2) bulkheads constructed with pressed salt blocks, (3) backfill composed of crushed salt material, and (4) backfill composed of earthen materials.

The design for the shaft sealing system in the SCP-CDR (Fluor, 1987c,d; see Section 5.2) is based, in entirety, on the schematic designs prepared for a generic site within the Permian Basin (Kelsall et al., 1985). This design addresses the key sealing issues for the shafts but does not contain the exact number and location of sealing components for each shaft and subsurface entry. These details were to be added during ACD and License Application Design activities when site characterization data are available. For the SCP design, the seal system schematics have been modified from the original designs to be compatible with the typical lined shaft. The schematic seal designs were in turn based on previous design studies (D'Appolonia, 1980a,b; 1981; 1982).

The philosophy behind the schematic designs is the use of short-term and long-term seals to create permeability and ground-water flow conditions similar to those of the intact rock. Short-term sealing is provided by concrete bulkheads. These seals are designed to prevent the intrusion of ground water to a lower strata until the long-term seals become effective. Long-term sealing is provided primarily by backfilling mined shaft material. However, the long-term sealing efficiency of earthen backfills may not be as good as for crushed salt backfills.

The schematic designs were developed from a preliminary design basis, described by Kelsall et al. (1985). This design basis can be satisfied by using dense earthen material mined from the respective strata as the major seal material to backfill shaft sections. Any additional backfill consolidation due to stress transfer from the adjacent rock would tend to improve the sealing of the earthen backfill. This approach is effective provided that (1) bulkhead components sufficiently retard the ground-water flow during the backfill consolidation period, (2) seal material is nearly insoluble near sources of ground water, and (3) seal component designs can sufficiently retard ground-water flow along the seal/host rock interface and through the construction-affected zone.

Sealing of deep boreholes was to be performed to decommissioning standards on those boreholes that are within the control area and are capable of transmitting aquifer ground water into the repository horizon. The aquifer waters in question are assumed to belong to the accessible environment. These boreholes were to be sealed according to strict sealing specifications.

Sealing of shallow boreholes that do not pose a threat to repository integrity was to be performed to less stringent oil field standards.

Underground boreholes that do not extend outside the host salt horizon (LSA-4 salt) will eventually seal through the natural salt creep process. However, they may be backfilled with crushed salt to accelerate sealing.

4.3.1 Accomplishments

The basic components of the shaft seal system are concrete bulkheads, structural backfill, dense earthen backfill, swelling clayfill, general earthen backfill, and the shaft cap (Roy et al., 1983).

Concrete bulkheads were to be the primary source of short-term sealing. These bulkheads were to be large, relatively impermeable structures that inhibit ground-water flow through the concrete, along the seal/host rock interface, and through the construction-affected zone. Precise locations of bulkheads will depend on site-specific considerations, including characteristics of candidate rock types, integrity of the formation, local ground-water conditions, and relative location to shaft lining components (e.g., bearing keys, operational seals, and type of lining). The total number of bulkheads required is currently undetermined because performance requirements and seal material design specifications have not been established.

Information obtained during each shaft sinking will be vital to developing final design of the shaft seal system.

Backfill sections must provide adequate long-term sealing against ground-water migration in excess of what the natural rock mass can transmit. To achieve this objective, the backfill material should be as similar to the surrounding rock as possible, recognizing, of course, that the material should consolidate to nearly initial in situ conditions and must be compatible with the local geochemical and hydrochemical environments. Seal materials were to be engineered to restore the disturbed excavation region to near its undisturbed state via swelling pressure. Critical long-term seal sections were to be in the lower salt strata, where backfill will have a very low permeability and the fracture healing behavior of the salt host rock will restore the disturbed excavation zone. The short-term seals between the aquifer and the lower salt strata must remain effective until the critical long-term seals become effective. Redundancy in the short-term seal design should prevent the failure of one seal component from compromising the entire sealing system. The upper seals were not assumed to be long-term seals. Eventual water seepage down to long-term seals at the salt strata does not appear to be a problem since only minimal leaching is expected before this water becomes saturated.

The critical lifetime for the short-term seals is several hundred to several thousand years, based on preliminary estimates of the time required for the backfill to consolidate to the lower permeability of intact salt. When consolidation of the backfill is sufficiently complete, the backfill will begin to function as a long-term seal. The long-term seals will prevent water flow downward into the repository horizon. The lifetime of the complete seal system must span the full 10,000-year period for isolation of the nuclear waste, independent of the short- or long-term designations.

The permeability requirements for the seal components were to be refined during the Seal System Component Investigations. Bulkhead seals are designed to be effective immediately or shortly after placement to limit ground-water flow through the bulkhead itself, along the interface between the bulkhead and the host rock, and through the construction-affected zone around the entry. Backfill seals, on the other hand, will require time to consolidate into a nearly impermeable material that retards ground-water flow through the sealed shafts.

The basic components of the borehole seal system are cement grout sections and clay sections.

Cement grout was to be used as the general sealing material because it is relatively easy to place, structurally competent, and highly resistant to ground-water flow. Pressure grout was to be used to squeeze grout into highly permeable rock zones, as necessary, to retard inflow from less competent, water-bearing strata. Provisions can be made to install reamed slots of cement grout in low-permeability, competent rock strata immediately below the major aquifers to intercept flow along the borehole-rock interface. This concept would be similar to the use of bulkheads in the shafts. The exact locations of each of these components will be determined primarily by the locations of the stratigraphic units and hydrologic basins.

Candidate materials for underground seals include crushed salt from the mining operations, nonsalt earthen backfills, cementitious concretes and grouts for backfills and bulkheads, and pressed salt blocks for panel bulkheads. Backfilling the underground workings with crushed salt during waste emplacement operations accounts for approximately 70 percent of the volume to be sealed. Crushed salt is the primary sealing material with respect to the long-term sealing objective and with respect to the volume of backfill material.

4.3.2 Planned Work

The Seal System Program was to be conducted through four specific programs: the Seal System Environment Program, the Seal System Components and Interaction Investigation Program, the Seal System Design Optimization Investigation Program, and the Seal System Modeling Program (ONWI, 1987c).

- The Seal System Environment Program included investigations needed to establish the repository seal and backfill environments. These investigations were to be performed to define the geotechnical characteristics that influence the design and performance of the seal system.

- The Seal System Components and Interaction Investigations Program included seal system component investigations and component-environment interaction testing. Repository backfill investigations also were to be performed in this program.

- The Seal System Design Optimization Investigation Program included seal system design optimization activities that were to require site characterization data. Activities included investigations to assist in design concept selection, development of design requirements, and studies to translate design requirements into specific design descriptions. Development tests to demonstrate feasibility of fabrication processes and to help verify the design were also included.

- The Seal System Modeling Program will include modeling studies associated with seal system development, utilization, verification, and validation for those investigations requiring data from site characterization. Activities will include development of seal component and subsystem models, the use of these models to conduct performance, safety, and optimization analyses, and tests to help assess the validity of these models.

Seal materials development consists of the development of three types of candidate seal materials: (1) salt materials, (2) cementitious materials, and (3) earthen materials.

Crushed salt is considered to be a prime candidate for use as a backfill material in the storage rooms and entry drifts and, possibly, in the lower portion of the shafts. Salt was also being considered as a sealing material in the form of preformed bricks or blocks. Activities related to the salt materials development included studies of (1) consolidation, (2) fracture-healing, (3) interface phenomena, and (4) emplacement methods.

The category of cementitious materials includes all the commercially available forms of these construction materials, including grout, shotcrete, mortar, concrete, and neat paste. The principal binder of cement in all of these materials is some form of Portland cement (a hydraulic calcium silicate cement). Activities for cementitious materials development included studies of (1) physical and chemical properties, (2) longevity and durability, (3) interface phenomena, and (4) emplacement methods (Coons et al., 1982).

The category of earthen materials includes crushed rock, gravel, sand, clays, and their mixtures. Clay-bearing materials can be tailored by appropriate selection of clay type and amount to have low hydraulic conductivity, high swelling capacity, high sorptivity, and high compressibility, all desirable properties for sealing applications. Also, clays are natural materials that are known to be stable in many geologic environments. Clay-bearing materials may be emplaced in a variety of ways, including conventional loose emplacement and in situ compaction, emplacement in precompacted blocks, and emplacement in a slurry form. The preconceptual designs propose the use of earthen materials as backfills in and near the shafts. Activities for earthen material development include studies of (1) physical and chemical properties, (2) longevity, (3) interface phenomena, and (4) emplacement methods.

Seal system design was being performed in several stages. For the Deaf Smith County site, the seal system design for the Permian Basin was to be updated as an advanced conceptual design and later as a license application design. The conceptual design was linked to quantitative design goals and bases developed by means of site and seal performance modeling.

In situ tests of seals and backfill have been identified for testing in the exploratory shaft facility. Three types of tests were to be performed: (1) a room seal test, (2) a room backfill test, and (3) a borehole seal test. The conceptual level details about these tests are included in the underground test plan (Golder, 1987). Detailed plans or site study plans were scheduled for development within the next couple of years.

Repository design was to occur in four sequential design phases. These phases are defined and described in detail in Section 4.4 of this report. Decommissioning seal design activities would also occur in parallel with these design phases.

4.4 FACILITY DESIGN

This element of work involves the engineering, design, and construction of all operating facilities at the repository, including on- and offsite surface and subsurface facilities. This activity did not include the decommissioning seal system design, which is included in Section 4.3. Over the period of time from conceptual design through license application, this element of work is concerned with those engineering and design activities required to develop design concepts and to prepare the detailed drawings and specifications required to license the repository and to support DOE budget submissions.

In addition, the Nuclear Waste Policy Act of 1982 requires the DOE to identify the scientific, engineering, and technical information required to site, license, construct, operate, and decommission a repository; estimate the costs for obtaining the data and information; and develop a schedule for obtaining the data and information. Repository operations and maintenance deals with those procedures and plans necessary to safely operate the repository. Repository decommissioning deals with the development of the plans and designs required to remove repository structures and facilities and restore the repository site to its original natural condition. At this stage of repository development, all repository operation, maintenance, and decommissioning activities involve evaluation of design-related features. For that reason, this report includes repository operation and decommissioning activities with repository design.

In the Mission Plan, the DOE established a four-phase iterative design strategy for repository development (DOE, 1985b, Section 2.2.2.1). These design phases are:

- Site Characterization Plan Conceptual Design. The objective of this phase is to support the salt repository site characterization plan. The site characterization plan conceptual design includes the repository conceptual design details necessary to understand site characterization information needs. Generally, this design phase emphasizes subsurface design because this is the area of the repository which defines site characterization program information needs. Surface and balance-of-plant preconceptual designs will also be developed to provide a general description of the operating repository.

- Advanced Conceptual Design. This design phase will complete repository design trade-off studies for all repository facilities and operations and select the concepts for detailed design. It will contain sufficiently detailed information to support congressional budget submissions and meet the requirements of various DOE orders for a major system acquisition project.

- License Application Design. This design phase will start with the concepts selected during the advanced conceptual design phase and will begin the detailed design process. The primary goals of this phase will be the resolution of design and licensing issues and the development of design details to the point that repository design compliance with Federal licensing regulations for a nuclear waste repository can be demonstrated. This design phase must develop sufficient engineering and design detail to support preparation of the safety analysis report, the environmental impact statement, the construction authorization license application, and the site selection report.

- Final Procurement and Construction Design. This design phase will be completed for the site selected for actual repository construction. This design phase will complete construction drawings and specifications. Generally, minimal engineering and design work will be required for systems, structures, and components that are important to repository site selection and licensing. This phase will be completed in parallel with the NRC review of the license application.

Additional design phases have not been specifically addressed in DOE program plans at this time. These plans will address design activities such as revisions to the FPC design due to construction impacts, preparation of "as-built" drawings, and input to the operating license application.

4.4.1 Accomplishments

Salt repository design activities were initiated in 1977 by Union Carbide Corporation for the Office of Waste Isolation (OWI) of the DOE. Two design studies were completed:

- National Waste Terminal Storage Repository Number 1 (NWTS-R1) (Stearns-Roger Engineering Company, 1979) is a study that developed a preconceptual design for a repository receiving canistered reprocessed waste and disposing of that waste in a hypothetical salt dome.

- National Waste Terminal Storage Repository in a Bedded Salt Formation for Spent Unreprocessed Fuel (NWTS-R2) (Kaiser Engineers, 1978) is a study that developed a preconceptual design for a repository receiving canistered spent fuel and disposing of that waste in a hypothetical bedded salt formation.

These studies were followed by a 1981 study, NWTS Conceptual Reference Repository Description (CRRD) (BGI, 1981). This study developed no new engineering design information; rather it compiled the Stearns-Roger NWTS-R1 and the Kaiser NWTS-R2 study results, developing a preconceptual design for a repository receiving canistered spent fuel and disposing of that waste in a hypothetical salt dome.

During 1982-83, Stearns-Catalytic (formerly Stearns-Roger) concluded a series of site-specific repository design feasibility studies for the seven candidate repository salt sites then under consideration (Stearns-Catalytic, 1987; Stearns-Roger, 1981, 1986). These studies

- Reviewed known site geologic and environmental data

- Developed site-specific surface and subsurface preconceptual designs

- Identified resource requirements (i.e., water, natural gas, electricity, etc.) for repository construction and operation

- Identified and evaluated transportation and utility access corridors

- Identified and evaluated possible disposal scenarios for excess mined salt

- Developed repository construction and operation source terms for evaluation of repository environmental impacts

- Developed repository construction, operation, and decommissioning schedules and cost estimates.

These studies described site-specific repository concepts with the capability to dispose of reactor spent fuel, reprocessed commercial and defense high-level waste, and transuranic waste. The Stearns-Catalytic design studies were used as the basis for the seven salt repository environmental assessments issued in May 1986 (DOE, 1986b). The repository defined by these studies included a waste-handling building that assembled the 300- to 1,000-year waste package required by 10 CFR Part 60. The previous studies did not include disposal in such a long-life package. In summary, this series of feasibility studies broadly defined the size and scope of a repository for disposing of nuclear waste in a salt host rock.

During 1985-87, Fluor prepared a conceptual design for the Deaf Smith County salt site to support the SCP (DOE, 1988). Specifically, Chapter 6 of the SCP contains a conceptual design of a repository, identifying likely site-specific repository issues. Results of the SCP conceptual design were documented in a separate, free-standing design report (Fluor, 1987d), which is a reference document for Chapter 6 of the SCP. The contents of the SCP conceptual design report were defined in an annotated outline (DOE, 1985a). From this document, the salt project requirements (SRPO, 1987a) and site-specific geotechnical data (DOE, 1986a), the scope of engineering work that was performed during this design phase was determined. The primary objective for this design phase was the identification of data needed for the site characterization program's repository design and design analysis (SRPO, 1987h). SCP conceptual design primarily identifies the area, extent, and depth of subsurface emplacement areas; the location and depth of repository shafts; the location and area of surface facilities; and the identification of site characterization data needed for repository design. Thus, the primary emphasis for the site characterization plan conceptual design is on subsurface and shaft design. Surface facility design data and information primarily define a framework for understanding the overall scope of a waste repository in a salt medium.

The reference repository conceptual design (Fluor, 1987d) is based on a vertical emplacement orientation for the individual waste packages. An alternative design report for horizontal emplacement has also been prepared (Fluor, 1987c). Emplacement orientation issues were to be further investigated during the advanced conceptual design phase. The different features of these two concepts are documented in a free-standing report (SRPO, 1987b).

The conceptual design also provided input to the plans described in Chapter 8 of the SCP to ensure that adequate information will be gathered to complete the remaining design phases. The conceptual design described in Chapter 6 of the SCP satisfies the requirement of the Nuclear Waste Policy Act of 1982, Section 113(b)(i)(c)ii.

In establishing the basic characteristics and configurations of the repository, the conceptual design presented in Chapter 6 of the SCP accomplished two purposes:

- Delineation of those structures, systems, components, and barriers important to safety and isolation that are necessary to receive, process, transport, and permanently dispose of radioactive waste in an underground facility, i.e., the Q-list

- Identification of needed information relative to both the design data base and the methods available for the engineering design of the repository.

Three overall capabilities must be considered in designing and operating the repository. The repository must be designed to safely emplace waste, retain the option to retrieve waste, and provide for the long-term containment and isolation of the waste.

Design elements of the repository for the Deaf Smith County (Texas) site include the following:

1. The main surface facilities. These were to be built on essentially flat terrain toward the northeastern end of the site. The surface facilities would occupy 600 acres and would be segregated into the waste receiving and inspection area, the waste processing area, and the general support facilities area (including excavated salt storage and treatment).

2. The shafts. Two shafts would initially be used for operation of the exploratory shaft facility. If the proposed repository is built, these shafts will be used as fresh air intakes for the waste emplacement area. The current design concept calls for the construction of four additional repository shafts: the waste shaft, the waste emplacement area exhaust shaft, the underground development area intake shaft, and the corresponding exhaust shaft.

3. Underground facilities. These were to be located in the Lower San Andres Unit 4 salt bed about 0.5 mi below the surface. The underground facility will occupy about 4,000 acres. It will include about 125 mi of emplacement entries and about 75 mi of access/ventilation entries.

In support of the SCP conceptual design, several engineering studies were prepared. Principal among these are:

- Waste package emplacement mode - comparison of horizontal and vertical orientations (SRPO, 1987b)

- Salt retrieval strategy - methods and development plans for retrieving waste packages after closure (Tomé et al., 1987)

- Shaft design guide - criteria for repository and exploratory shaft design (SRPO, 1987g)

- Salt minimization - evaluation of design changes and resulting impacts that would occur if the repository concept were optimized to reduce the amount of salt excavated during repository operation (SRPO, 1987c)

- Sensitivity analysis - impact of variations in key design parameters on repository design and its effect on site characterization program (SRPO, 1987d)

- Rod consolidation - impact of consolidating spent fuel assemblies on cost and schedule of repository system (SRPO, 1987e)

- Subsurface HVAC requirements - literature survey of existing mine designs vis-a-vis proposed gassy mine regulations to develop preferred ventilation criteria for repository subsurface (SRPO, 1987f)

- Repository Q-list - a preliminary Q-list (systems, components, and structures important to safety and waste isolation) based on a probabilistic risk assessment of repository subsurface operations (SRPO, 1988a).

These and other engineering studies (SRPO, 1987i through z; Fluor, 1987a,b) provided support to the SCP conceptual design report.

4.4.2 Planned Work

A long term program plan for the repository work element includes a strategy for future design efforts (ONWI, 1987a). Based on the SCP conceptual design, the waste package conceptual design, and the subsequent performance assessment (see Section 4.5.), the repository advanced conceptual design can be performed. According to the Mission Plan, "The advanced conceptual design phase will be used to explore design alternatives and will firmly fix and refine the design criteria and concepts to be made final in later design efforts. The project feasibility will be demonstrated, life-cycle cost estimated, preliminary drawings prepared, and a construction schedule developed as required in DOE Order 6410.1" (DOE, 1985b, p. 221).

The license application (LA) design is the third phase in the phased repository design approach. According to the Mission Plan, "The LA design phase will complete the resolution of design and licensing issues identified and assessed in earlier design phases and will develop the design of the items necessary to demonstrate compliance with the design requirements and performance objectives of 10 CFR Part 60. Therefore, sufficient design information will be developed during the LA design phase to meet the requirements of 10 CFR 60.31 for the license application. Design requirements resulting from detailed safety and reliability analyses will be fully integrated into the LA design and will form the basis for information required in the safety analysis report" (DOE, 1985b, p. 222). The LA design would incorporate in situ test data from the exploratory shaft facility to support the LA.

The final procurement and construction design is the fourth and final phase in the phased-design approach. According to the Mission Plan, "This design phase will emphasize the completion of design of ancillary support items, final design refinement for the items necessary to demonstrate compliance with the design criteria and performance objectives of 10 CFR Part 60, the development of construction bid packages for all systems, and the development of final construction and procurement schedules. Minimal disruption in the NRC review process will be experienced during this design phase since the level of design detail on safety-related systems and components will have been adequately covered in the LA design" (DOE, 1985b, p. 222).

During each design phase, it was intended to integrate the design input/output with other program elements, e.g., transportation (ONWI, 1987b).

4.5 PERFORMANCE ASSESSMENT

The scope of this element of work is to mathematically model the waste containment and isolation performance of the decommissioned and sealed repository subsurface (i.e., the near field). A major portion of this work involves the predictive modeling of water flow through the seal system portion of the engineered waste isolation barrier.

Repository postclosure performance assessment activities were to assemble the design results, models, data, and information, and to develop a model of the repository near field that would predict the performance of the repository engineered structures and components. In effect, these analyses would estimate the fluid flow rate through the decommissioned repository structures and components over future geologic time.

The results of the repository performance assessment were to be incorporated into the total system performance assessment (see Chapter 1, Section 1.4).

4.5.1 Accomplishments

Performance assessment modeling activities include acquisition, development, verification, and validation of computer codes in the following areas:

- Thermal effects
- Thermomechanical effects
- Fluid flow analysis
- Radionuclide transport analysis.

Model development of thermal codes is nearly complete. The principal remaining activity is the development of a quality assurance (QA) traceable benchmarking for these codes so that they may be accepted by the NRC for repository licensing. The principal thermal codes used in postclosure performance assessment were to be TEMP, SPECTROM-41, STEALTH, And HEATING6. Additional code development work was to involve code sensitivity and uncertainty analysis. Additional details on future thermal code and model development activities are shown in Section 8.3.5 of the SCP (DOE, 1988). Code verification and validation have been ongoing activities. Latest efforts are focused on code review, QA documentation, and revision of codes.

Thermomechanical modeling of salt and nonsalt materials involves a more difficult technical challenge than the development of thermal models. Thus, thermomechanical codes and model development have lagged behind the development of heat transfer models. Currently, the STEALTH codes represent the most fully developed program for repository postclosure thermomechanical performance assessment. Final revision and documentation of the first SPECTROM-32 version has been completed. Further development for thermomechanical codes depends on completion of constitutive models for the stress-strain behavior of salt and nonsalt materials. Specific details on the development of thermomechanical codes and models are described in Section 8.3.5 of the SCP (DOE, 1988). As with thermal codes, verification and validation of thermomechanical codes is ongoing.

Given the very low permeability of rock salt, no significant fluid flows are expected through the undisturbed rock salt units of the near field. Fluid flow analysis will take into account the properties of the backfill systems (both the consolidated salt backfill and the earthen backfill materials), the decommissioning seal systems (including cementitious bulkheads), the time-dependent properties of the disturbed salt rock units, and the stratigraphy of surrounding rock units.

Recent developments in the Federal Republic of Germany and by other investigators have prompted the consideration of moisture movement through salt in the vapor phase as well as in the liquid phase. SUTRA is the only readily available unsaturated flow code under consideration for repository postclosure performance assessment. Of the codes available for analysis of saturated flow, only FED3DGW and SWENT are ready for Hydrocoin Level II validation problems for calculation of 3-dimensional saturated ground-water flow. Postclosure performance assessment plans have been made to upgrade CFEST to allow 3-D calculation of moisture movement through unsaturated media and to include a correction for thermal expansion of pore fluid.

Aside from the continued development of sophisticated finite-element and difference codes, repository postclosure performance assessment activity was developing the SYSNET code to analyze the combined effects of salt creep and salt dissolutioning from intrusion or connection events originated by human activity. A system code, TUSC, was also under development and was to be used to model fluid flow across the entire repository system.

Recent evidence indicates that moisture movement through salt has little to do with brine inclusion migration under a thermal gradient. Consequently, new constitutive models for fluid flow through salt were being developed, particularly for combined fluid-phase and vapor-phase moisture movement. Lack of data from field or rock mechanics laboratory testing on such properties as hydraulic conductivity as a function of vapor pressure or degree of saturation has delayed the ultimate verification and validation of new fluid flow models in salt. In addition, a new model is being developed to relate joint/fracture apertures in the salt and nonsalt rock units to hydraulic conductivity in the near field. This model will couple thermomechanical and fluid flow assessment for intact and crushed salt, as well as the nonsalt units. Analyses using this model will require defining the hydrologic flow from the far field to the near field.

Radionuclide transport codes and models for repository postclosure performance assessment vary from simple analytical solutions, to 1-, 2-, and 3-D numerical solutions and gressed versions for sensitivity uncertainty analyses. Most of the available codes are operational but many need additional development.

Verification and validation of radionuclide transport models continues. Analytical solutions and benchmark comparisons will continue to verify numerical transport codes such as SWENT, CFEST, and SUTRA. In addition, numerical codes such as CFEST are being used to verify simple submodels such as those used for the calculation of radionuclide diffusion along a moisture concentration gradient. Plans have not yet been developed for the systematic validation of various components of radionuclide transport codes with either repository rock mechanics laboratory testing or field in situ testing in the exploratory shaft. Analysis of radionuclide transport to date has not generally included radionuclide retardation. Therefore, coupling transport with geochemical codes such as EQ3/EQ6 has not been a concern. This position was to be reexamined as site-specific data on clay seams, interbeds, etc. in the host rock unit were obtained from the site characterization program.

As mentioned in Facility Design (Section 4.4), postclosure performance assessment was to be completed after each design phase. Work was completed on the SCP conceptual design performance assessment, based on existing codes and models (ONWI, 1987d,f). As discussed in Section 4.4.1 (page 4-20), conceptual designs were developed for both horizontal and vertical emplacement modes. Although the horizontal design was used for the analysis to support the design review, waste package emplacement orientation had no effect on the results or conclusion because of the lack of detailed site-specific data and the preliminary nature of both the design and the codes/models used. This analysis identified potential concerns in repository design performance. These concerns were to be addressed during the advanced conceptual design phase and were to be used as input to future updates of the SCP.

4.5.2 Planned Work

A performance assessment plan was to be completed and issued. Model development and verification/validation for the thermal, thermomechanical, fluid flow, and radionuclide transport models was to continue. As repository design phases were completed, successive performance assessments were to be performed.

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DOE ORDERS

DOE Order 6410.1, Management of Construction Projects.

CHAPTER 5

REGULATORY AND INSTITUTIONAL (WBS 5.0)

From the beginning of considering the disposal of high-level nuclear waste in a mined geological repository, it was apparent that a regulatory process such as that used by the U.S. Atomic Energy Commission or, later, the U.S. Nuclear Regulatory Commission would be imposed. The U.S. Department of Energy, early in the high-level nuclear waste program, emphasized to its contractors the importance of conducting activities in a manner that would comply with regulatory requirements and be supportable in the licensing process. These activities are presented in three sections of this chapter: licensing, environmental compliance, and institutional.

5.1 LICENSING

As part of a National Energy Plan, President Carter ordered a review of the U.S. Nuclear Waste Management Program. An internal U.S. Department of Energy (DOE) task force carried out the review and published a draft report in February 1978. The report highlighted the need to develop a national nuclear waste management policy and integrated program that incorporated the views of the involved government agencies, the Congress, the States, local governments, industry, the scientific and technical community, and other members of the public. On March 13, 1978, in response to the task force findings, the President established the Interagency Review Group on Nuclear Waste Management (IRG) to formulate recommendations for an Administrative policy with respect to the long-term management of nuclear wastes and supporting programs to implement the policy.

The Interagency Review Group released its draft report for public review and comment on October 19, 1978. Approximately 3,300 comments from every one of the fifty States were received and reviewed. About three-fourths came from private individuals. A substantial number were from State governments, the utility industry, the nuclear industry, public interest organizations, environmental organizations, academia, and the general business community. Responses were also received from Congress and a number of Federal agencies.

Comments were given careful consideration and were used by the Interagency Review Group in formulating its recommendations to the President. The Interagency Review Group's report was issued in March 1979 (IRG, 1979).

The report showed that the primary objective of the DOE's waste management planning was in compliance with the primary objective of the U.S. Nuclear Regulatory Commission's regulatory program for radioactive waste disposal. It then remained for the Interagency Review Group to recommend the facilities that should be subject to licensing. In considering this question, the IRG identified the following three principles as relevant to the matter of which DOE nuclear waste facilities should be licensed:

- National security guarantee. Consistent with past expressions of Congressional intent, no DOE facility should be regulated by an outside authority if such regulation would potentially inhibit the production of materials for national defense or lead to the disclosure of national security information.

- Equivalent protection. The extent of the public's exposure to nuclear waste materials does not vary by ownership of the facility or origin of the material. Thus the public must be assured equivalent protection from material from both government and nongovernment sources.

- Independent regulation. In the area of nuclear energy, the public is best served by independent regulation consistent with national security guarantees. The Congress clearly intended this in the Energy Reorganization Act of 1974.

The Interagency Review Group considered the following three alternative options to define the degree of licensing coverage appropriate for DOE facilities:

1. The status quo, as contained in existing legislation.

2. An extension of NRC licensing authority (requiring new legislation) primarily to incorporate licensing of new DOE facilities for disposal of transuranic (TRU) waste and nondefense low-level waste (LLW) or

3. A further extension of NRC licensing authority to incorporate all new DOE post-reprocessing waste facilities and interim storage facilities as well as disposal of waste from both the defense and nondefense programs (DOE, 1978).

The Interagency Review Group agreed to recommend alternative 2 and described it in detail as set forth below:

- Commercially Generated Spent Fuel. Commercially generated discarded spent fuel is not presently considered to be nuclear waste. Nevertheless, the disposal of such material present hazards similar to those encountered in the disposal of high-level waste (HLW), and therefore should be licensed. The NRC could designate commercially generated discarded spent fuel as HLW through rulemaking. Alternatively, specific legislation could require the licensing of any disposal in DOE facilities of such material. In any event, any DOE-owned new away-from-reactor (AFR) facility used to store commercially generated spent fuel should be licensed, since AFR storage is a substitute for commercial storage, which is subject to licensing. Legislation will be required.

- TRU Waste. Although disposal of TRU waste generated from military activities is not now subject to licensing, it should be since the permanent disposal of such material presents long-term hazards comparable to those encountered in the disposal of HLW, which is licensed. Legislation will be required.

- Uranium Mill Tailings. Legislation now before Congress redefines "byproduct material" to include radon and radium in uranium mill tailings. If the legislation is approved, new mill tailings would be generated and managed under NRC regulatory control.

- Intermediate-Scale Facilities (ISF). If an ISF is to be used for permanent disposal of spent fuel, HLW, or TRU waste, that facility should be subject to licensing. Legislation will be required.

- New Nondefense LLW Disposal Sites. Commercially controlled shallow land burial of LLW is already subject to licensing. Existing LLW burial sites operated by DOE are associated with the defense program and should continue to be exempt from licensing. However, if DOE acquired any existing commercial LLW site or opened a new nondefense LLW site, such sites should be subject to licensing. Legislation will be required.

In April 1979, the U.S. Department of Energy released the Final Environmental Impact Statement on the Management of Commercially Generated Radioactive Waste (DOE, 1980a). This EIS in concert with the Report to the President by the Interagency Review Group on Nuclear Waste Management (IRG, 1979) recommended that a mined geologic repository be the preferred first disposal facility.

The status of repository program work and technology in 1979 is described in three key reports: Report on Geologic Exploration Activities (ONWI, 1980b), Earth Science Technical Plan... (ONWI and USGS, 1980), and Status of Technology... (Klingsberg and Duguid, 1980).

In February 1980, the President of the United States, in his Message to Congress, "adopted an interim planning strategy focused on the use of mined geologic repositories."

The Proposed Rulemaking on the Storage and Disposal of Nuclear Waste (Waste Confidence Rulemaking) resulted in a Statement of Position of the United States Department of Energy released in April 1980 (DOE, 1980b). The position concluded that there was no technological impediment to implementation of high-level waste disposal in geologic repositories. The chief concerns noted were the ability of NRC and EPA to promulgate the needed regulations and standards and for NRC to establish the licensing process to ensure compliance with the implementation schedule.

Finally, in January 1983, the Nuclear Waste Policy Act of 1982 became law and the process and schedule for selecting and developing a mined geologic repository was established.

5.1.1 Early Licensing Tasks

Battelle's Office of Nuclear Waste Isolation (ONWI) performed many of the early licensing tasks for the DOE using subcontractors; Bechtel and NUS were

Regulatory Program Managers. The Licensing Project Manager (LPM) contract was placed with Ebasco in 1982. These and other subcontractors accomplished the following:

- Prepared a Conceptual Reference Repository Design (CRRD) (BGI, 1981) using forerunner designs by Stearns-Roger and Kaiser Engineers for bedded and domal salt formations. The reference design was used to produce the Preliminary Information Report, intended to be an early tool for discussing licensing issues with NRC.

- Prepared a multivolume Preliminary Information Report (PIR) and associated multivolume Environmental Report (ONWI-121) (ONWI, 1981) on a reference repository site and the CRRD. The report included site and design descriptions, preclosure and postclosure safety and performance assessments, environmental effects, and environmental monitoring. The performance assessment results in the PIR were supported by work from the WISAP and AEGIS programs (see Chapter 1) from Battelle Pacific Northwest Laboratories.

- Issued a proposed Regulatory Guide (in the context of U.S. Nuclear Regulatory Commission Regulatory Guides) for the Standard Format and Content of a Safety Analysis Report for a repository--1982

- Developed a Licensing Plan and licensing topical reports

- Published a Site Characterization Report Preparation Guide--1982

- Wrote a draft Site Characterization Report--1983

- Developed networks for producing either
  - one site characterization plan (SCP), or
  - one SCP for each salt site.

- Prepared an Annotated Outline for the Preparation of an Environmental Report for a Geologic Repository, November 1981 (ONWTSI, 1981).

5.1.2 Planning

The promulgation of the Nuclear Waste Policy Act of 1982 caused a reordering of site characterization planning activities in order to comply with the procedures, schedule, and nomenclature of the Act.

The Activity Plan for Preparation of the Site Characterization Plan (SRPO, 1987b) was prepared. As the process or emphasis of SCP activities changed, the Activity Plan was modified to stay current.

A Site Characterization Plan Preparation Specification (SRPO, 1985) was produced to provide, in one document, all of the background and guidance that SCP writers would need. It incorporated the regulations, regulatory specifications, issues hierarchy, issue resolution strategy, and glossary. The specification was superseded by an annotated outline discussed below.

The guidance of U.S. NRC Regulatory Guide 4.17, Standard Format and Content for a Site Characterization Plan (NRC, 1984), was used as a basis for the annotated outline, but was expanded to provide more detailed direction for SCP writers. After review and revision within DOE-OCRWM, the annotated outline was presented to the NRC staff and discussed. DOE baselined the Annotated Outline for Site Characterization Plans (DOE, 1985, 1987) for use in the program.

5.1.3 SCP Preparation

The preparation of the SCP was a major SRP undertaking. The main activities involved in the SCP preparation are discussed below.

Information Sheets--An initial step in SCP preparation was to prepare an approved base of data for use by all writers of the plan. The mechanism selected was that of preparing information sheets identifying the descriptor being quantified, rationale for the quantification, and references supporting the choices. Information sheets were prepared for the disciplines of Geotechnology, Environment, Waste Package, and Engineering Design. The latter category consisted of geotechnical information that was processed and provided early as a basis for exploratory shaft design. These sheets were titled "Synthetic Reference Data," because they were synthesized from information available at the time. The sheets were assembled into a document entitled Synthetic Geotechnical Reference Data for the Deaf Smith Site (DOE, 1986).

All of the information sheets that were prepared were given technical review, revised as necessary, and submitted for baselining. Revisions to approved or baselined information sheets were made as the need was identified. Revised sheets were submitted to the same review and baselining regimen.

Readiness Review--A review of the Salt Repository Project's readiness to write and issue the Site Characterization Plan was directed by DOE-SRPO in August 1986. Following this review, ONWI was instructed to initiate preparation of the SCP.

Reports of the status of completing readiness review activities--closing open items and eliminating technical holds--were prepared biweekly initially. As the work proceeded with no problems, the reporting frequency diminished to monthly and then bimonthly. The closeout report was submitted and accepted in February 1988.

Writers' Briefing--The initial contact of SRP-SCP participants was a writers' briefing in Columbus, December 9, 1986. The principal writers or their managers were briefed on the means to be employed to direct and control the content and style of the SCP and to determine the schedule status and appropriateness of their writings. More than 140 writers or managers attended the one-day briefing and left with a schedule for future actions.

Narrative Outline--The writers' first assignment was to produce a narrative outline of their allocated portions of the SCP. This outline was to be an expansion of the baselined Annotated Outline indicating the depth to which each topic would be pursued. The figures and tables were to be identified, the references to be relied on cited, and the SCP interfaces or internal references listed. This information was provided by each writer, and then the composite narrative outline was reviewed by ONWI for completeness and overall compatibility. Review comments were returned to the writers for consideration.

Author Workshop--A workshop involving all of the SCP authors was held in Columbus from February 9 to March 6, 1987. The initial meetings of the workshop drew approximately 200 writers. The following tasks were accomplished:

- Resolution of comments on the narrative outline
- Revision of the narrative outline in accordance with the resolution of comments
- Identification of internal interfaces, agreement on their treatment, and approval by all parties (storyboard process).

Writing SCP--The SCP participants left the Author Workshop with an approved narrative outline and authorization to proceed with writing the SCP. Schedules varied depending on the schedule of the coordinating ONWI department. Writing progress status was reported at least weekly at the instigation of Licensing Department coordinators. Writing progress was assessed directly by site visits of Licensing Department and, where applicable, Quality Assurance staff of ONWI.

Reviewing SCP--Each ONWI Department responsible for a chapter of the SCP arranged for it to receive a technical review (the review of Chapter 6 was arranged by the Repository Department). In addition to technical merit, several other review categories were required by the Licensing Department, such as:

QA considerations	Quality Assurance
Legal considerations	Legal
System compatibility	Systems Engineering
Performance modeling requirements	Performance Assessment
Reference usage	Production Services

On completion of the ONWI review and revision, the chapters were delivered to the DOE's Salt Repository Project Office (SRPO).

Onsite Review--DOE-Headquarters' onsite reviews were conducted in Columbus, Ohio, and Hereford, Texas. Chapters and sections were sent by HQ or by DOE-SRPO to the designated HQ reviewers. The onsite reviews were convened for two to four days to resolve comments and agree on the associated text changes.

After each of the initial onsite reviews, the marked-up text was sent to Battelle Publication Services to be revised and reprinted for further review. According to the schedule, Chapter Reviews at HQ, to ascertain the efficacy of the resolution of onsite review comments, was to have been the next step.

Chapter 8--The completion of certain sections of Chapter 8 was on a slower schedule than the rest of the SCP. With understanding performance allocation requirements, preparing issue resolution strategies, and determining the content of investigations, this stretched schedule was not unusual. The DOE-Headquarters dictated a schedule change requiring completion of a consultation draft SCP by January 8, 1988. With this compression of the anticipated schedule, there was not time for the normal progression of the form and content of Sections 8.2, 8.3, and 8.5. Section 8.2 had been completed and reviewed by ONWI. The comments were received but had not been reconciled with the text. Section 8.3 had been completed and was about to be submitted for review. Section 8.5 only existed in conceptual outline drafts.

Chapter 8 Workshop--In order to meet the accelerated schedule demands, a workshop was held in Amarillo to rewrite Sections 8.2, 8.3, and 8.5, and to re-review introductory Sections 8.0 and 8.1.

Two weeks of intensive effort resulted in rewriting of Sections 8.2, 8.3, and 8.5 to the joint satisfaction of DOE-Headquarters, DOE-SRPO, and the Salt Repository Project participants.

The master, marked-up text of these sections was sent to Publication Services to be revised and reprinted and combined with the previously completed chapter and sections.

Assembled Document Review Workshop--In another concentrated two-week session, this time in Washington, comments from the review of the assembled document were resolved and text changes were proposed and agreed to. Master, marked-up text was again processed through Publication Services to obtain a concurrence draft for DOE-HQ signoff.

Concurrence Review--The process of obtaining DOE-HQ concurrence for the printing of the SCP was addressed by DOE-SRPO management and staff supported by ONWI. After the first week of review, authorization was received to print Volumes 1 and 4. By Tuesday of the second week, additional authorization was received and by weekend, total concurrence was received.

Printing--The Consultation Draft SCP was to have been printed and distributed widely for the consultation process in January 1988. The final volumes were printed and all sets were quarantined in the Battelle warehouse awaiting future direction from the U.S. Department of Energy on their distribution or other disposition.

5.2 ENVIRONMENTAL COMPLIANCE

The Salt Repository Project (SRP) was required to comply with all applicable statutes, rules, regulations and other legal authorities that did not conflict with the requirements of the Nuclear Waste Policy Act of 1982. Executive Order (EO) 12088, Federal Compliance with Pollution Control Standards (1978); DOE Order 5400.1, and the Environmental Policy Statement in the Site Suitability Guidelines (10 CFR 960.5-2-5(b)(1)), requires that the SRP identify and make provisions for compliance with all applicable procedural and substantive environmental requirements.

These requirements are extensive and include a host of statutes, regulations, executive orders, and other directives. Compliance with these procedural and substantive requirements can take a variety of different approaches. For example, in some cases all that is required is that DOE demonstrate a "good faith" effort to achieve its regulatory responsibilities. In other cases, a particular permit, letter, or other form of authorization may be required from a Federal or State agency. Finally, environmental compliance may require a major programmatic effort; for example, compliance with the National Environmental Policy Act (NEPA) requires a complex procedural and documentation process culminating in the preparation of an Environmental Impact Statement.

In addition to these non-program-specific requirements, the Nuclear Waste Policy Act of 1982 (PL 97-425) established additional requirements or reinforced other regulatory requirements. Specifically the Act required the Department of Energy to:

- Monitor for and mitigate to the extent practicable significant adverse environmental and social impact if conditions indicated the need (Section 113(a))

- Prepare a plan for decommissioning of the site if found unsuitable for construction authorization, and reclamation of the site and mitigation of significant adverse impacts caused by site characterization activities (Section 113(b)(1)(A)(iii) and (c)(4))

- Evaluate the site suitability for the location of a repository in terms of the criteria developed pursuant to Section 112(a) (Section 113(b)(1)(A)(iv))

- Develop a Consultation and Cooperation Agreement with affected States and Indian Tribes (Sections 111(a)(6) and (b)(3), 113(a) and (b), 116(c)(2)(A), and 117 (b))

- Make available to the public and submit to the President a final environmental impact statement prepared pursuant to NEPA, as specified in the NWPA, in making a recommendation for construction of a repository (Section 114(a)(1)(D))

- Plan for and conduct environment investigations during the site characterization phase that will sustain the granting of a construction authorization by the U.S. Nuclear Regulatory Commission (Section 114(B)).

The following sections describe the SRP Environmental Compliance Program, discuss the Environment Compliance Support of the Site Selection Process, address the work on the Environmental Assessments for the seven salt sites, and discuss the activities of the Environmental Compliance Program following the completion of the Environmental Assessments.

5.2.1 Description of Environmental Compliance Program

The SRP Environmental Compliance Program provided compliance support to SRPO. This support included the following activities:

1. Provide management and technical direction on environmental compliance.

2. Identify, catalog, and maintain the baselined set of technical and administrative project requirements that derive from the SRP objectives of:

   - Preserving environmental quality
   - Efficient and effective use of resources
   - Public participation
   - Compliance with applicable statutes, codes, and standards
   - Achieving technical excellence.

3. Develop SRP-specific interpretations of policy and strategies that identify preferred approaches among competing options for the fulfilling of functional requirements related to environmental compliance.

4. Identify or develop for all project functional requirements related to environmental compliance performance measures and standards that will represent acceptable levels of achievement.

5. Identify the required data and information to support the acquisition of approvals, authorizations, and registration for the conduct of SRP field work.

6. Document compliance through the preparation of reports that detail proposed action of the results of SRP actions.

7. Establish and maintain systems that track project progress in satisfying environmental program compliance and documentation needs.

8. Establish SRP-wide strategies for the incorporation of statutory compliance procedures and requirements into site characterization, design, construction, and performance assessment.

9. Support implementation of the National Environmental Policy Act (NEPA) including the preparation of implementation plans and environmental statements.

10. Provide review of proposed facilities siting, construction, operation, and decommissioning to assure compliance with environmental requirements.

11. Support the Salt Repository Project Office in tri-project Statutory Compliance Coordination Group interactions.

12. Provide staff support to the Technical Evaluation Department and the Operations Department as appropriate to conduct technical review of their results.

5.2.1.2 Environmental Compliance Activities Prior to the Passage of the NWPA (Approximately 1978 until December 1982)

5.2.1.2.1 Generic Environmental Impact Statement

The Final Environmental Impact Statement: Management of Commercially Generated Radioactive Waste (DOE, 1980a) evaluated nine different options for the disposal of high-level and transuranic waste:

1. Isolation in deep, conventionally mined underground repositories

2. Placement in very deep drilled holes

3. Placement in underground cavities in such a manner that radioactive decay would lead to rock melting

4. Island disposal in mined repositories

5. Subseabed disposal in sedimentary deposits

6. Ice sheet melting

7. Deep well injection as a grout slurry

8. Partitioning of the waste and transmutation of long-lived radionuclides

9. Disposal in space using the space shuttle.

Of these, it was determined that disposal underground in deep underground repositories could be made available most quickly and did not involve unacceptable environmental effects.

5.2.1.2.2 Environment Compliance Support of the Site Selection Process

Regulatory compliance concerns were an integral part of the environmental and geologic site identification programs (see Section 3.1). The ability to license the project site through the NRC and obtain the necessary environmental permits was always a consideration in the selection of the site. For example, early on, when the program was still considering extensive geographic area, the status of land that was protected by environmental regulations or as a matter of national policy (e.g., park land and wilderness areas) was considered in the site screening process.

Both the environmental and geologic portions of the site selection process were guided by a set of ten siting criteria, Nuclear Waste Terminal Storage Program Criteria for Mined Geologic Disposal of Nuclear Waste: Site Performance Criteria (DOE, 1981a). Although these criteria were established in the absence of specific guidance from the regulatory agencies, they were developed in consideration of the then proposed NRC and EPA regulations.

The first step in selecting a nuclear waste repository site was the identification of criteria against which a site can be judged. Although it was ultimately the responsibility of the U.S. Nuclear Regulatory Commission using standards to be established by the U.S. Environmental Protection Agency to determine what constitutes an acceptable site, the Department of Energy is the agency responsible for the selection of the site.

The Nuclear Regulatory Commission, on February 25, 1981, published regulations for licensing geologic repositories for the disposal of high-level waste (NRC, 1981). These regulations, however, contained only general siting and licensing provisions. In 1982, the Nuclear Regulatory Commission published Technical Criteria for Regulating Geologic Disposal of High Level Radioactive Waste (NRC, 1982). These criteria addressed those performance factors related to public health and safety but did not propose regulations covering environmental and socioeconomic factors. The Nuclear Waste Policy Act of 1982 (PL 97-425) required the Department of Energy to establish, in conjunction with the NRC and concerned states, a set of guidelines against which a proposed repository site could be judged. However, these siting guidelines were not finalized until after the site selection process was completed.

Because the NRC regulations were under development throughout most of the process of identifying the first candidate sites, the DOE developed a set of ten siting criteria (DOE, 1981a). These provided a basis for the characterization and selection of sites. They included criteria that are important to the containment and isolation of the waste as well as factors that determine the environmental and socioeconomic acceptability of a site and are directed towards the dual objectives of protecting public health and safety and protecting the environment. These criteria were based on criteria defined for geologic repositories by the National Academy of Sciences (NAS, 1978) and the International Atomic Energy Agency (IAEA, 1977); siting criteria used during earlier nuclear waste programs (Brunton and McClain, 1977; DOE, 1980b); and guidance from the Nuclear Regulatory Commission (NRC, 1980).

The criteria broadly and in general terms specified conditions related to physical properties, physical phenomena, and potential social, economic, and environmental considerations. The criteria were hierarchical in character, and precedence in siting was given to health- and safety-related factors over environmental and socioeconomic factors.

These criteria were used within the context of a screening process described in the National Plan for Siting a High-Level Radioactive Waste Repository and Environmental Assessment (DOE, 1982a) and summarized in Chapter 3 of this report.

5.2.2 Environmental Assessments for the Salt Sites

A major activity between 1983 and 1986 was the preparation of the Environmental Assessments for the seven salt sites, as required by the Nuclear Waste Policy Act of 1982. The Environmental Assessment for each nominated site was required to contain the following:

- A detailed statement of the basis for the site nomination and probable impact of the site characterization activities
- An evaluation of the impact of detailed geologic, environmental, and socioeconomic studies on public health, safety, and the environment
- A discussion of alternative activities to avoid impacts associated with site studies
- An assessment of the regional and local impacts of locating a repository at the site
- An evaluation of site suitability under the siting guidelines, to be based on the available information base and not require the development of additional information
- A description of the site selection process
- A reasonable comparative evaluation of the site being described in the environmental assessment in relation to other sites being considered.

Draft Environmental Assessments were prepared for seven salt sites: the Deaf Smith and Swisher sites in the Permian Basin (DOE, 1984a,b), the Davis Canyon and Lavender Canyon sites in the Paradox Basin in southeastern Utah (DOE, 1984c,d), and Vacherie, Richton, and Cypress Creek dome sites in Mississippi and Louisiana (DOE, 1984e,f,g).

The preparation of these Environmental Assessments was the main focus of the Environmental Compliance Program from 1983 through the end of 1984 when they were made available for public and State review on December 20, 1984.

Through a public and State information process and as a result of a 60-day review period, over 13,000 comments were received on the salt site Environmental Assessments. The number of comments and the development of the required responses delayed issuing the final Environmental Assessments until May 28, 1986.

The NWPA required that the environmental assessments be finalized to accompany the nomination of at least five sites. Through a DOE decision analysis process, the Deaf Smith, Richton, and Davis Canyon salt sites, along with the Hanford and Nevada sites, were nominated with final EA's completed.

5.2.3 Environmental Compliance Program Since the Completion of the Environmental Assessments

The most recent Environmental Compliance Program was developed following the issuance of the Environmental Assessments. The program was influenced by three factors: the completion of the Environmental Assessments, which allowed the program to focus on the site characterization phase; the establishment of an independent environmental compliance program with a well-defined agenda; and the development of a Systems Engineering Management Plan as a management tool for guiding the program (see Section 1.1.1).

The Systems Engineering Management Plan (SEMP) for the Salt Repository Project (SRPO, 1986) identified four types of documents: requirements documents (i.e., what ought to be done), strategy documents (i.e., what approach will be used), implementation plans (i.e., how it will be done), and organizational plans (i.e., who will do it). One of the key documents required by the SEMP was the SRP Requirements Document.

The Salt Repository Project Requirements Document (SRPO, 1987a) translated the generic program requirements into detailed program requirements (Section 1.2.4). It consisted of Functional Requirements, specific statements of the basis operations that must be performed by the SRP-MGDS; Performance Criteria, the level of performance it must achieve and the means of assessing that performance; Constraints, limitations imposed, environmental conditions within which the program must function; and Assumptions.

A systematic approach was required for the Salt Repository Project to address all of the regulatory needs as they pertain to transportation, environmental, socioeconomic, and radiological health and safety matters during all phases of repository development. In order to fulfill this responsibility and the intent of the Salt Repository Project Requirements Document (SRP-RD), the Environmental Compliance Program developed the IRS process. The Issue Resolution Strategy (IRS) process involved five steps that systematically identified activities and information needs necessary to fulfill the SRP-RD Functional Requirements and Performance Criteria. These five steps, which start with the SRP-RD, are briefly described as follows:

Step 1. Identify program functional requirements, performance criteria, constraints, and assumptions through the Systems Engineering Management Plan process.

Step 2. Identify a set of issues or question from the Requirements Document (RD) which, when resolved, will demonstrate compliance with the program requirements identified in Step 1.

Step 3. Develop a strategy or approach for the resolution of each of these issues. The strategy will consist of a systematic sequence of steps which, when accomplished, will satisfactorily resolve the issue. Included in this approach will be the development of criteria or standards to be applied in determining if the issue is satisfactorily resolved.

Step 4. Identify, develop, and defend the choice of a particular assessment methodology or standard to be used in the implementation of the resolution strategy.

Step 5. Identify the information necessary to implement the strategy and resolve the issue. This information list then would be handed off to the field contractor to support the implementation of the field work.

This approach when completed would result in environmental compliance program specifications that would guide the environmental, socioeconomic, and radiological health and safety study programs. In addition this approach would identify how and where the information/data and summary analysis received from the field, laboratory, or library would be used to resolve specific issues and contribute to the documentation of major findings. This approach would also provide the basis for tracking the resolution of an issue in a Site Investigation Status Report (SISR). This would be possible because all information needs would be tied through the issue to specific sections and subsections of the RD, and to a particular phase of the program. In order to facilitate the development of the environmental, socioeconomic, and transportation Site Study Plans (SSPs) and support the collection of field and laboratory data, all information needs would be ordered by discipline and identified as to appropriate program phase.

5.3 INSTITUTIONAL PROGRAM

The SRP institutional program had its origins in the series of recommendations made in March 1979 to President Jimmy Carter by the Interagency Review Group on Nuclear Waste Management (IRG, 1979). In its final report the IRG recommended mechanisms for cooperating with State, local, and Indian Nation officials as well as mechanisms for increased public participation in the government's waste management program. Even earlier, in 1978, the essential purposes of institutional activities were scoped in a public information plan. In its published form, in May 1979, the ONWI Public Information Plan (ONWI, 1979a) provided guidelines "to allow for the preparation and dissemination of factual information in order to enhance understanding by federal, state, and local officials and thereby encourage informed participation in the program by citizens and governments."

As the program matured through the years, its focus was defined and narrowed from a national scope to more specific foci in regions, areas, and sites being considered for a nuclear waste repository. At the same time the program evolved from one concentrating on information dissemination to a program stressing participation in planning and, where possible, decision making by State and local groups and private citizens. A process to develop a public participation plan was devised to determine information needs and desired types of interaction.

Activities for carrying out this institutional program are described in the following paragraphs. Most of these were implemented by ONWI acting in support of the DOE's Salt Repository Project Office (SRPO). As with other parts of the SRP, SRPO reported to the DOE's Chicago Operations Office in carrying out the policies of the Office of Civilian Radioactive Waste Management (OCRWM) at DOE Headquarters.

5.3.1 Institutional Tasks Prior to the Nuclear Waste Policy Act of 1982

Public Information Plans: The IRG report (IRG, 1979) called for three types of activities in planning nuclear waste disposal:

- Technical activities, resting on well-founded scientific and engineering bases
- Management that is comprehensive enough to integrate all elements and provide for evaluation of achievements against expectations
- Institutional activities which are open to wide and diverse participation, are flexible enough to accommodate changing preconceptions, and are sensitive to people's concerns.

To implement the institutional recommendations, DOE's Richland Operations Office (which at that time was the contracting office for ONWI) published a public information plan (ONWI, 1979a) which established an open information policy; provided for public meetings and for the distribution of informative materials; and stressed the need for coordinating the program closely with members of Congress and with units of State and local government. This plan was short and very general. Subsequently, it was modified for the Chicago Operations Office of DOE when that office became responsible for administering the program. However, the plan provided basic guidance for DOE's national institutional planning.

In 1982, a series of six draft public information five-year plans was prepared for the projects which, at that time, made up the National Waste Terminal Storage Program.

Materials Development: The various public information materials originally developed by ONWI and ONI for DOE use in the late 1970s and the 1980s have been modified and augmented by DOE. A number of these materials continue to be utilized in national DOE public information activities. For example, a series of twelve fact sheets on nuclear waste topics first prepared in 1979 are still in use (ONWI, 1979b).

An early large-scale materials development effort was the preparation in 1979 of a nine-projector multi-image slide presentation for the State Planning Council on Radioactive Waste Management, a group of governors and state officials which served during the last years of the Carter administration. Later dubbed "The Challenge of Nuclear Waste Isolation," this presentation was updated several times and used widely in two-projector and videotape forms through the mid-1980s. "Nuclear Waste Isolation, A Progress Report," a 16-mm film, was produced for ONWI in 1980 (ONWI, 1980a). During the 3 years it was circulated nationally by DOE, it was seen by more than five million people through schools and other group showings and via cable television.

In 1980 ONWI contracted with the Columbus, Ohio, science museum for the construction of two large-scale science museum exhibits. (Originally, the intention was to circulate one of the exhibits in the eastern United States and the other in the west.) These exhibits, titled "Nuclear Waste Isolation, Challenge of the 80s," were designed to facilitate viewer participation with interactive computer displays, radiation and shielding demonstrations, and actual samples of a dummy spent fuel element and spent fuel cask. The exhibits were previewed in Columbus at the 1981 NWTS national information meeting. The two exhibits were shown in eight major science museums in 1982, their only full year of use. In 1983 portions of the "eastern" exhibit were made available by DOE to various OCRWM projects. The "western" exhibit was divided between ONWI's Hereford, Texas, and Moab, Utah, information offices, where it remained until the offices closed.

A number of other specific information products were developed to support DOE's concept of "consultation and cooperation" with State and local governments. State briefing books on each of the four States with salt sites were important tools for DOE and contractor staff members in interacting with representatives of State and local groups and other Federal agencies. Updated frequently, each book included information on a given State and site area, together with telephone numbers of key contacts, discussions of ongoing field activities, and a chronology of major DOE interactions with the State. Special-purpose briefing books also were used frequently at meetings between DOE and State representatives. State mailing lists were maintained and updated, and both published reports and unpublished field data were made available to States. With nomination of the Deaf Smith County site in 1986, additional site-specific information materials were created.

<u>Public Oversight Through the Program Review Committee</u>: The concept of "peer review" of ONWI's institutional activities was incorporated into the repository project through the Program Review Committee (PRC), a distinguished group of leaders and scholars who were not otherwise associated with the nuclear waste program. Headed by a retired college president, the PRC consisted of 12 to 13 members who were nominated by national professional organizations and learned societies. Meeting in various project locations approximately six or more times a year, starting in 1980, the PRC devoted significant time to enhancing interactions with State and local groups and to reviewing the institutional implications of important documents and plans such

as the draft Environmental Assessments for the salt sites in Louisiana, Mississippi, Texas, and Utah.

Nomination of the Deaf Smith County, Texas, site in 1986 marked a change in scope for the PRC as it did for other parts of the institutional program. At the time of project cancellation in late 1987, significant progress had been made in reconstituting the PRC so that it more closely reflected the interests and concerns of those living in the Southwest, particularly in Texas.

Annual Meetings of the National Waste Terminal Storage Program: To promote the exchange of information from its high-level nuclear waste program, both within the program and with other groups, DOE in 1979 began a series of five annual national meetings. These meetings were open to the public and were attended by contractors, university scientists and engineers, representatives of State and local governments and Federal agencies, pressure group representatives, and residents of areas near potential nuclear waste repository sites. Annual meeting attendance rose from several hundred in 1979 to nearly three thousand people in 1983. Detailed proceedings were published, summarizing the state-of-the-art of knowledge of nuclear waste isolation (DOE, 1979; 1980c; 1981b; 1982b; 1984h). Introducing the fifth annual meeting, in Washington, DC, in December 1983, Deputy Secretary of Energy Daniel Boggs commented on DOE's planned institutional interactions (DOE, 1984h):

> Interactions are not only mandated by statute but are desired by the Department, by the Secretary, and by all of us to involve you in this process in a way that is useful, meaningful, and important to all the interests that you represent. This is a very clear and strong commitment of the Department of Energy, and I am sure you will see that carried through by the many DOE representatives that will be working with you in the days ahead.

5.3.2 Fulfilling NWPA Institutional Requirements

DOE's Salt Repository Project Public Information Offices: Development of public information offices in the four salt states provided some of the most tangible evidence of the SRP's institutional approach to repository siting. The first information offices were established in July and August of 1982 at Monticello and Moab, Utah, under sponsorship of the Utah Office of Nuclear Waste. After the Governor of Utah replaced the state Office of Nuclear Waste with a Policy Steering Group as liaison with DOE, in December 1982, DOE itself (on January 3, 1983) assumed responsibility for the two offices. On August 4, 1983, the mayor of Minden, Louisiana, requested to DOE that an information exchange be held as soon as possible and that a public information office be opened. Suitable space was located, and on June 14, 1984, the office in Minden opened its doors. This preceded by little more than a month the opening of an information office in Richton, Mississippi, on June 21. The first two Texas information offices were opened February 1, 1985, at Tulia and Hereford. The information office at Vega, Texas, opened November 1, 1985. In November 1987, the Tulia information office was closed, and arrangements were made for moving its reference materials to an information office in Amarillo which would have opened in early 1988.

Each of the information offices was staffed nearly full time by a professional trained in library science. Each office provided a local center for technical reference materials on the project as well as a source of printed materials, DOE and contractor news announcements, small exhibits, and audio-visual materials. Information office staff members had telephone access to DOE and contractor staff members who could provide factual answers to questions asked by local citizens. Information offices were centrally located in each community and also provided limited facilities for small group meetings and one-on-one contacts between project personnel and local residents. Although the nuclear waste project was a matter of public controversy in most local communities, office staff were trained to provide information in a neutral, non-advocacy context. Literature from groups opposed to nuclear waste repositories was made available at the offices when the groups requested distribution space. Throughout the project there were no reported incidences of violence or vandalism at any of the information offices.

Consultation and Cooperation With State and Local Officials: Quoting from the Nuclear Waste Policy Act of 1982, Section 117 (b): "In performing any study of any area within a State...and in subsequently developing and loading any repository within such State, the Secretary shall consult and cooperate with the Governor and legislature of such State...to resolve the concerns of such State...regarding public health and safety, environmental, and economic impacts of any such repository."

The program enlisted government affairs consultants in Louisiana, Mississippi, and Texas and replied, formally, to all letters it received as part of the consultation and concurrence process. The Salt Repository Project maintained a consultation and cooperation file and kept a chronology of significant interactions. There were 43 such interactions for Louisiana from 1978 through 1982, 70 for Mississippi, 80 for Texas, and 117 for Utah. From 1983 (following passage of the Nuclear Waste Policy Act) through 1986 there were 156 such interactions for Louisiana, 260 for Mississippi, 309 for Texas, and 302 for Utah. As a record of consultation and cooperation, and to provide information for DOE and its contractors, an extensive newspaper and magazine clipping file was maintained and compilations of clippings on nuclear waste were circulated at frequent intervals.

Highlights of consultation and cooperation activities during the 1983-1986 period included the following:

- Public hearings on the proposed Siting Guidelines, held in five cities in March 1983
- Public hearings on the proposed nomination of salt sites for detailed characterization, held in the four states in April and May 1983
- Eight bimonthly meetings of salt state representatives and SRP staff in Columbus, Ohio, during 1983 and 1984 to discuss technical criteria for siting a repository and other issues of common interest

- News conferences, briefings, and availability sessions in Amarillo, Tulia, and Hereford, Texas, in March 1984, to explain the identification of possible study sites in the Texas Panhandle

- Briefings and hearings on the final environmental assessments for the Richton, Mississippi; Deaf Smith County, Texas; and Davis Canyon, Utah, sites to salt state governors' representatives, May 28, 1986. Delivery of the EAs and supporting documents on the following day to information offices, local officials, and others in the salt states.

Activities Supporting the EA Reviews and Dissemination: In supporting public reviews of the draft and final salt site EAs, a three-step approach was employed. This involved, first, assisting State and local groups and individuals in finding information within each EA that was pertinent to their needs and interests. A series of information-exchange sessions in local communities was held to provide this assistance. Typically, a session would consist of an introduction to the general content of the EA by Headquarters and DOE Salt Repository Project Office (SRPO) staff followed by an availability session where DOE and contractors provided information on specific parts of the EA on a one-to-one contact basis. Simultaneously, the EAs, supporting documents, and other information were made available through libraries and through DOE information offices. Ultimately, each information office maintained a full collection of EA supporting documentation. Services of a speakers bureau also were made available.

In the second step, several days after the information-exchange sessions, formal DOE hearings were held in several salt state locations. Testimony during the hearings was utilized in preparing the final EAs, and comments were addressed in a separate comment response document for the three salt EAs that were edited into final form.

Finally, care was taken to deliver the final EAs, including the decision to characterize sites in Nevada, Texas, and Washington, to governors' representatives of all States being considered for the first repository at the same time. This delivery was followed one day later by delivery to other interested groups.

Public Participation Plan Development: As siting activities became more specific, public participation planning for States and localities was intensified. In January 1984, SRPO approved a public participation plan. In March 1985, ONWI signed a contract with Susan Wiltshire, a nationally recognized public participation consultant. The first draft of a proposed SRP public participation planning process document (see SRPO, 1987c) was prepared and reviewed prior to nomination of the Deaf Smith County site in 1986.

5.3.3 Activities Since Nomination of the Deaf Smith County Site

Project Relocation: From an institutional standpoint, preparations were in order for relocating the SRP to the Texas Panhandle shortly after the Deaf Smith County site was nominated for detailed characterization on May 28, 1986.

Information offices in Texas were stocked with reference materials, exhibits, and program literature. Contacts had been made with State and local officials, although there was no formal consultation and cooperation agreement as envisioned by the NWPA. Public participation planning was well along. Essentially, there was a stalemate of several months while other elements of the nuclear waste program began to prepare site characterization plans for Nevada, Texas, and Washington.

On January 14, 1987, DOE announced that a six-month process of relocating personnel to Texas would begin in February. Subsequently, on February 9, DOE announced that major SRPO and contractor offices would be located in Hereford and Amarillo. To provide details of staff relocation, and to describe upcoming site characterization activities, DOE-SRPO held an extensive series of news media briefings and public meetings in the Panhandle during the week of February 23-27. Briefings also were held for local officials in Deaf Smith, Potter, Randall, and Oldham counties in Amarillo, Canyon, Hereford, and Vega. On March 2, SRPO and ONWI opened temporary offices in trailers in Vega, to respond to questions on job opportunities, contracts, and permanent office locations. Several meetings were held with business and real estate groups to discuss relocating some 300 DOE and contractors (largely Battelle) employees to offices in Hereford. Community and business leaders and educators from the Panhandle took part in three meetings in Columbus, Ohio, for relocating staff members and their families.

As relocating staff arrived, the Vega trailers were overcrowded quickly. On May 5, DOE/SRPO and two contractors moved into temporary office facilities in Hereford. On May 27, the SRPO deputy manager began weekly meetings in Hereford (which lasted through November 4) to provide status reports on site transition, study planning, hiring, procurements, and outreach activities. DOE/SRPO, ONWI, and Parsons-Redpath hosted open houses, June 20, in the Hereford offices. ONWI institutional staff supported a SRPO contracting opportunities workshop in Hereford on July 16. By October 1, institutional staff had completed their project relocation.

Community and State Interactions: The February 23-27, 1987, meetings in Texas began an intense period of State and community interactions in the Panhandle. The program responded to numerous requests for information by the Texas Nuclear Waste Programs Office and the Waste Deposit Impact Committee (WDIC), a community group in Hereford funded by DOE through grants to the State of Texas, and other State and local groups. A speakers' bureau complied with more than 100 requests for speakers for local groups during the year. Exhibits were shown at meetings of the Society of Exploration Geophysicists (Dallas, March) and Texas Library Association (San Antonio, April), and at the Tri-State Fair (Amarillo, September) and the Panhandle South Plains Fair (Lubbock, September/October). On September 22, the SRP Land Acquisition Plan (SRPO, 1987d) was mailed, for review, to owners of land within the nine-square-mile site and to owners of potentially affected parcels outside the site.

Public Participation Plan Finalization: Development of a process for planning public participation in the site area was well under way before the Environmental Assessments were finalized. A first draft of the proposed public participation process document was transmitted to SRPO for DOE review on

October 21, 1985. On June 13, 1986, a seminar was held for SRPO and ONWI managers to increase their awareness of public participation planning and of the importance of early public involvement in project activities. In July, meetings were held to identify early opportunities for public participation in technical activities. The concept was introduced to the Panhandle in topical presentations during the February 23-27, 1987, public meetings.

May 4, the final draft of the public participation planning process document was transmitted to SRPO. Planning began with a series of working sessions for Panhandle residents May 5-7 and meetings with project technical staff in June and July. During July and August, speakers bureau presentations were made to several area groups, and volunteers were recruited for future planning activities. During the last week of September, there were six informal workshops for Hereford and Vega residents to learn more about site characterization.

Following submission of the annotated outline on November 5, the draft SRPO State, Local and Public Participation Plan (SRPO, 1987c) was submitted to DOE at the end of December 1987.

Planning for Site Characterization: In August 1987, a new process was proposed by DOE Headquarters for review of Site Characterization Plans (SCPs). Instead of the planned 90-day public comment period, the SCP would be issued, simultaneously with SCPs for the other two repository projects, in relatively early draft form for initial consultations with the Sate of Texas and the NRC. Consultation workshops with State agencies and the NRC would be held in the first three months of 1988. During this period, a full set of SCP references would be available for review. The Environmental and Socioeconomic Mitigation Plans (MMPs) would be released concurrently to complete a "total picture" a site characterization. Publication of a revised draft SCP and the full 90-day public review and hearing period would then follow.

Institutional planning for the SCP "consultation draft" was organized to support release of the draft on or about January 8, 1988. Because the SCP was planned to be a large and complex document, it was necessary to develop a guide for potential reviewers. The Public Handbook for the Site Characterization Plan for the Deaf Smith County Site in Texas (DOE, 1988) was rushed to completion in December 1987. Plans were made for training Texas information office staff in helping people identify various parts of the SCP and for providing SCP references in the Texas information offices.

5.3.4 Institutional Activities in Conjunction With Project Closure

Four SRP information offices, in Richton, Mississippi; Moab, Utah; and Hereford and Vega, Texas, remained open at the end of 1987. Closeout plans for the Richton and Moab offices were accelerated so the two offices could close February 29, 1988. Library reference materials were made available to local education institutions, and audiovisual materials and exhibits were shipped to Hereford for disposal. The Vega office was scheduled to be closed March 31, 1988. However, it was anticipated that the Hereford office would remain useful as a community contact point until the SRP was no longer in the Panhandle, after September 30, 1988. Institutional records, audiovisual materials, literature, and exhibits were inventoried before turnover to appropriate DOE organizations. It was suggested that many of the materials would be useful to the Nevada project, to OCRWM Headquarters, and to the DOE transportation program.

An important institutional task was to maintain close contacts with local officials as well as with concerned citizens and the media as long as there was an SRPO presence in Texas. Institutional staff continued to support DOE by drafting letters in response to inquiries by concerned members of the public and by State and local groups such as the Waste Deposit Impact Committee (WDIC is a quasi-governmental group representing socioeconomic interests in Deaf Smith County). Speakers' bureau commitments were honored through March 1988, although speakers concentrated on the generic aspects of the nuclear waste program rather than on aspects specifically relating to the salt sites.

Institutional support was made available for interacting with State and community groups affected by the reclamation of remaining SRPO field sites in Louisiana, Mississippi, Texas, and Utah.

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CHAPTER 6

EXPLORATORY SHAFT (WBS 6.0)

6.1 BACKGROUND

The purpose of the exploratory shaft facility (ESF) was to gain access to the potential repository horizon and perform in situ tests for characterization of the site to determine acceptability for a repository. The shafts also circulate fresh air to the subsurface operations and contain conveyances for material or personnel handling requirements. Specific relationship of the ESF to the repository and its role in testing is described in the Site Characterization Plan (SCP) (DOE, 1988) and Site Characterization Plan Conceptual Design Report (SCP-CDR) (Fluor, 1987a,b).

Historically, the present ESF design evolved in the course of several steps. First, seven potential sites for a nuclear waste repository in salt were identified and initial preliminary designs were completed for the three sites of highest potential (Parsons Brinckerhoff/PB-KBB, 1985), and a review of these designs was made by ONWI (1986a). The Deaf Smith County site in the Permian Basin was selected for further evaluation, and a final preliminary design was prepared (Parsons Brinckerhoff/PB-KBB, 1986). After the Environmental Assessment (EA) was issued (DOE, 1986a), a Review of Readiness to perform final (Title II) design was performed by ONWI for DOE and items were identified which needed resolution either before or during Title II design (SRPO, 1986). During the development of the Title II design, three extensive reviews were made at the 30%, 60%, and 90% completion levels to determine if the design was satisfying the criteria (ONWI, 1987a, b; 1988).

The criteria under which the ESF design was developed are contained in several project documents. Preliminary design was performed using functional design criteria (ONWI, 1986b). Title II design was performed using criteria from the Salt Repository Project Requirements Document (SRPO, 1987c) and in particular Section 1.1.3, the exploratory shaft facility portion of the document. Using this basis, Parsons Brinckerhoff/PB-KBB prepared supplemental design criteria (Parsons Brinckerhoff/PB-KBB, 1987c). Additional criteria are also provided in the Shaft Design Guide (SDG) (Fluor and Parsons Brinckerhoff/PB-KBB, 1987a), Input to Seismic Design (ISD) (Fluor and Parsons Brinckerhoff/PB-KBB, 1987b), Testing Interface Specification (TIS) (Golder, 1987c), Draft Underground Test Plan (Golder, 1987e), Shaft Study Plan (Golder, 1987d), ESF Flexibility Position Paper (SRPO, 1987a), Site Population Study for ESF (ONWI, 1987d), and the Synthetic Geotechnical Design Reference Data for the Deaf Smith County Site (DOE, 1986b), among others. These criteria are implemented in the design through the procedures established in the project Quality Assurance (QA) Operating Procedures (Parsons Brinckerhoff/PB-KBB, 1987b) developed to implement SRPO's quality program as stated in Quality Assurance Specification to Parsons Brinckerhoff/PB-KBB for Activities in Support of Design of the Exploratory Shaft Facility (SRPO, 1987b). This assures that the design is properly managed, documented, verified, and traceable.

The following sections of this chapter provide information on two major studies and a summary description of the ESF at the time of closeout of the Salt Repository Project.

6.2 SHAFT DESIGN GUIDE

The geological regime at the Deaf Smith County, Texas, repository site indicates the shafts were to be constructed through horizontally bedded sedimentary deposits consisting of unconsolidated and weakly cemented soils, variable-strength rocks, and evaporite sequences. The soils include uncemented and unindurated cohesive or noncohesive material, such as the loess of the Blackwater Draw Formation, and the silty sand and gravel of the Ogallala Formation. Two major aquifers, known as the Ogallala Formation and Dockum Group, occur at the site. In addition, salt strata of various thicknesses exist in the Upper Seven Rivers, Upper San Andres, and Lower San Andres (five units) formations. Unit 4 of the Lower San Andres Formation had been selected as the repository horizon.

Competent rocks may not require any external support, but unconsolidated material or weakly cemented rocks may require support of the rocks. Safety in both type rocks depends upon controlling water inflows especially in water-soluble evaporites such as rock salt and potash. The support of rocks and control of water are key functions of the shaft lining and of the seals constructed at strategic locations.

The purpose of the Shaft Design Guide (SDG) (Fluor and Parsons Brinckerhoff/PB-KBB, 1987a) was to provide a common approach to design both the repository and exploratory shafts planned for the Salt Repository Project (SRP) repository in Deaf Smith County, Texas.

This shaft design guide is the first to take into account the following requirements, which exceed those for conventional shaft design:

1. The nominal design life must be 100 years.

2. Seismic effects on design must be considered.

3. Excavation methods must minimize overbreak and disturbance of the host rock.

4. Design and construction must allow for the installation of decommissioning seals.

5. Monitoring requirements and concerns for confirmation of shaft performance must be defined.

The Mission Plan for the Civilian Radioactive Waste Management (CRWM) Program (DOE, 1984a) specifies:

The DOE intends to use the exploratory shafts, as required, to ensure that the construction of the repository can be completed in time to meet the January 1998 date mandated by the act and will continue to evaluate the most cost-effective use of the exploratory shafts in the operating repository.

Appendix E to the Generic Requirements for a Mined Geologic Disposal System (GRMGDS) (DOE, 1984b) implemented the cited Mission Plan intention by requiring exploratory shafts that can be incorporated into the repository and can be used to support repository construction. Another requirement from Appendix E of the GRMGDS mandates that exploratory shaft design and construction methods demonstrate constructability for the candidate repository shafts. Successful implementation of these programmatic requirements required that a common design approach be employed by both exploratory and repository shaft designers, and the SDG was used to achieve this.

The SDG is supplemented by data bases including the SRP data base (DOE, 1986b) and the Input to Seismic Design (ISD) (Fluor and Parsons Brinckerhoff/PB-KBB, 1987b). The data in these supporting documents are specific to the Deaf Smith County, Texas, repository site. The SRP Data Base contains the stratigraphic, hydrogeological, geochemical, geomechanical, and thermal design data. The ISD contains information on the seismic stratigraphy and material properties, seismic assessments, seismic design parameters, and host media stability analyses.

The SDG addresses the design of vertical circular shafts requiring both watertight and nonwatertight lining sections. These shafts would accommodate waste handling, mined salt handling, intake and exhaust ventilation, and service access. The SDG describes the following components and requirements that influence the shaft lining design: operational seals, shaft bottom plug, and station areas; surrounding rock and soil structures; attachments to the liners; instrumentation and monitoring requirements; and decommissioning bulkhead requirements. The SDG required the designer to:

- Incorporate information from the reference design data bases
- Use the types of linings and seals identified in the SDG as the bases for selecting suitable shaft lining structures
- Use the methods and forumlae identified in the SDG as the bases for calculating loads, stresses, and displacements of lining and seal components
- Calculate lining dimensions using closed-form solutions and/or computer analysis methods described in the SDG or comparable methods
- Identify monitoring requirements for performance confirmation of the shaft design.

The SDG does not address the design of ground freezing systems or hoisting systems including headframes, hoists, guides, hoist ropes, and conveyances.

The SDG does indicate how loads from these items should be considered in the shaft design, however.

The identification of specific licensing requirements for the SRP shafts was beyond the scope of the SDG. Licensing guidance was to be provided by subsequent SRP documents. The shaft systems, structures, and components important to safety and the engineered barrier systems important to waste isolation were to be identified by the shaft designer following approved Salt Repository Project Office procedures.

The exploratory and repository shaft designs were to be in accordance with approved QA procedures.

6.3 HOIST STUDY

The hoist study (ONWI, 1987c) was made to develop a project position and recommendation on the type, size, and procurement approach to use in order to provide the hoisting capability at the ESF site during development, test support, and decommissioning.

The hoist study showed a preference for purchasing hoists over any option involving lease or rental costs. Given the planned duration of site characterization, government purchase should yield significantly lower total costs for the project.

Double-drum, double-clutch, 1,000-HP hoists are preferable to single-drum configurations and somewhat better than double-drum, single-clutch. While the purchase cost for this option is somewhat higher than the others, the benefits from increased versatility, minimum project schedule impact, and improved productivity seemed to favor the double clutch. This choice is favored for both construction and operations phases of the project.

Newly fabricated or suitable unused available hoists are favored over used hoists based on the reduced risks of breakdown, easier maintenance, warranty, startup assistance availability, and quality assurance. While new hoists cost more to purchase and require longer to procure, new hoists rank above used hoists, and project milestones provided adequate time for procurement.

Technical suitability, flexibility, safety, cost, and schedule were considered in the study. However, the scope of the study was limited to support of site characterization activities, and support of repository construction was not considered.

Alternate decision analysis matrices were developed for each recommendation to examine the decision's sensitivity to different subjective viewpoints. None of the recommendations were found to be sensitive to minor changes in rankings against or weightings of the various decision criteria.

6.4 ESF SURFACE FACILITIES

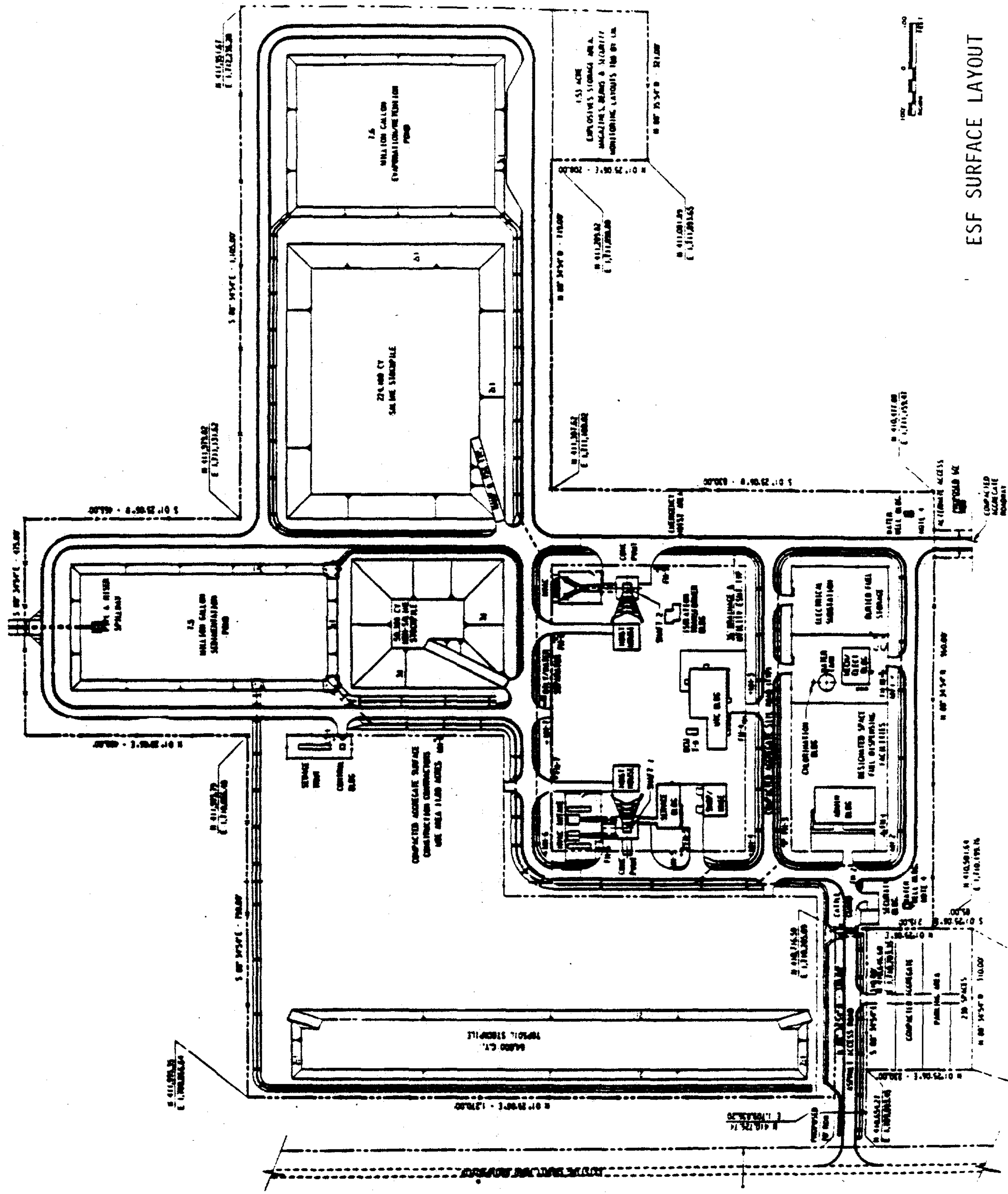
The surface facilities (buildings) at the ESF necessary to support site characterization are shown in the site layout (Figure 6-1) and include the following:

- Security building
- Administration building
- Mechanical/electrical building
- Chlorination building
- UTC building
- Shop/warehouse
- Service building
- Isolation transformer building
- Sewage treatment plant control building
- Shaft 1 and 2 hoist houses
- Shaft 1 and 2 shaft houses.

A summary of major ESF building characteristics is given in Table 6-1. In general, the preferred structures for the ESF were pre-engineered metal buildings because of short delivery time, simplified erection, and flexible interior space arrangement. Exceptions are the mechanical/electrical building, a portion of the UTC building, and the isolation transformer building, which were reinforced concrete to be tornado resistant. The mechanical/electrical and isolation transformer buildings were tornado resistant to ensure standby electrical power supply to either operational hoist, mine ventilation system, fire pumps, critical testing equipment and instrumentation, mine dewatering system, communication systems, hazardous gas detection systems, and compressed air to meet minimum operational requirements. The computer equipment portion of the UTC building was designed for tornado resistance to insure against loss or interruption of site characterization and test data transmitted from the shafts and subsurface test area.

Shafts 1 and 2 had identical pre-engineered metal building hoist houses. Each hoist building provided a control room for the hoist operator, an electrical control center with environmental isolation consisting of filtered air and positive pressure, the hoist, and restroom facilities. The shaft house control room included video monitoring of the shaft collar area.

The Shaft 1 hoist was designed as the principal service hoist for everyday transport of personnel, material, equipment, and supplies between the surface and the subsurface station. It consisted of a double-drum, double-clutch, direct-current (DC) motor-driven hoist that operated a personnel/materials cage in balance with a counterweight. The capacity of the cage was approximately 30 people or a 6-ton equipment load with provisions for suspending below the cage any equipment too large to fit inside. The Shaft 2 hoist was designed as an alternate personnel hoist for evacuation of the subsurface, as a means of occasionally hoisting broken rock to the surface, and as a means of shaft access for instrument monitoring, shaft inspection, and maintenance. It consisted of a double-drum DC motor-driven hoist that operated a cage in an



ESF SURFACE LAYOUT

Table 6-1. ESF BUILDING CHARACTERISTICS

BUILDING	FLOOR SPACE	FUNCTION	UNIQUE EQUIPMENT/FACILITIES	STRUCTURE TYPE
SECURITY	2,400 ft^2	PROVIDE EMERGENCY AND SECURITY SERVICES	AMBULANCE, FIRE TRUCK, FOAM TRAILER, SURFACE AND MINE SAFETY MONITORING/ALARM SYSTEM	PRE-ENGINEERED METAL BUILDING
ADMINISTRATION	9,020 ft^2	FACILITIES FOR GOVERNMENT SPONSORED PERSONNEL AND ESF DOCUMENTS DURING CONSTRUCTION AND TESTING AND PERSONNEL TRAINING	MULTI-PURPOSE ROOM FOR 50 VISITORS, CONFERENCE OR TRAINING USA	PRE-ENGINEERED METAL BUILDING
MECHANICAL/ELECTRICAL	4,940 ft^2	ENCLOSURE FOR ESF MECHANICAL AND ELECTRICAL EQUIPMENT	3 - 900 SCFM AIR COMPRESSORS 2 - 1200 KW DIESEL DRIVEN GENERATORS 2 - 1500 GPM FIRE WATER PUMPS 2 - 100 GPM AND 1 - 50 GPM POTABLE WATER PUMPS PRIMARY DISTRIBUTION SWITCHGEAR	REINFORCED CONCRETE, TORNADO RESISTANT
CHLORINATION	24 ft^2	ENCLOSURE FOR GAS CHLORINATION OF POTABLE WATER FOR THE ESF	50 LBS/DAY CHLORINATOR 6 - 150 LBS CHLORINE GAS CYLINDERS	FIBERGLASS, ONE PIECE MOLDED BUILDING
UTC	11,407 ft^2	HOUSES THE ADMINISTRATIVE AND COMPUTER AREAS FOR TESTING INSTRUMENTATION AND DATA COLLECTION UNIT ASSEMBLY	SURFACE ADAS CENTER, DCU TRAILER ASSEMBLY, GEOLOGY FIELD LAB	PRE-ENGINEERED METAL BUILDING WITH REINFORCED CONCRETE COMPUTER CENTER, TORNADO RESISTANT
SHOP/WAREHOUSE	3,500 ft^2	FACILITIES ARE ROUTINE MAINTENANCE AND REPAIR OF EQUIPMENT AND GROUNDS WITHIN THE FACILITY AND SAFE STORAGE AND DISPENSING OF MATERIALS	HYDRAULIC HOSE SWAGER, WELDER, GRINDER, DRILL PRESS, CONDUIT BENDER, HYDRAULIC PRESS, ETC.	PRE-ENGINEERED METAL BUILDING
SERVICE	4,000 ft^2	HOUSES SHOWER, LOCKER AND MINE RESCUE EQUIPMENT FOR PERSONNEL WORKING UNDERGROUND	ENCLOSED AIRLOCK/WALKWAY TO SHAFT HOUSE NO. 1, MINE RESCUE TRAINING ROOM, PERSONNEL MINE EQUIPMENT DISPENSING AND MAINTENANCE/REPAIR	PRE-ENGINEERED METAL BUILDING
ISOLATION TRANSFORMER	704 ft^2	HOUSES ISOLATION TRANSFORMER AND SWITCHGEAR	2,000 KVA TRANSFORMER, UNDERGROUND FEEDER SWITCHGEAR, NEUTRAL GROUND RESISTOR	REINFORCED CONCRETE, TORNADO RESISTANT
SEWAGE TREATMENT CONTROL	114 ft^2	HOUSES SEWAGE TREATMENT PLANT CONTROL PANEL AND RECORDERS, PROVIDE LABORATORY FACILITIES TO MONITOR PLANT PERFORMANCE	FLOW RECORDERS, SMALL SCALE WATER QUALITY LABORATORY	PRE-ENGINEERED METAL BUILDING
SHAFT 1 & 2 HOIST HOUSES	3,136 ft^2	HOUSES THE SYSTEM FOR TANSPORATION OF PERSONNEL, MATERIALS AND EQUIPMENT BETWEEN THE SURFACE AND SUBSURFACE	1 - 1,000 HP DOUBLE-DRUM/DOUBLE CLUTCH HOIST EACH HOIST HOUSE, OPERATOR CONTROL ROOM WITH CCTV OF SHAFT COLLAR AREA	PRE-ENGINEERED METAL BUILDING
SHAFT 1 & 2 SHAFT HOUSES	2,116 ft^2 SHAFT 1 2,479 ft^2 SHAFT 2	PROVIDE SAFE AND CONTROLLED ACCESS FOR PERSONNEL, MATERIALS, VENTILLATION, UTILITIES AND EQUIPMENT FROM THE SURFACE TO THE UNDERGROUND TEST FACILITY	46 ft^2 SERVICE CAGE SHAFT 1 2 - 15.9 ft^2 CAGES SHAFT 2 AIRLOCK SYSTEM BOTH SHAFT HOUSES MINE HAVC INTAKE SHAFT 1, EXHAUST SHAFT 2	PRE-ENGINEERED METAL BUILDING
TOTAL	45,504 ft^2			

SOURCE: PARSONS BRINCKERHOFF / PB-KBB, 1987 a unbalanced mode. The cage was mounted in a guide frame and accommodated 15 people for emergency evacuation at a design rope speed of 1,200 ft/min.

Additional surface facilities necessary to support the ESF included the following:

- 250,000-gal potable/fire water above ground steel storage tank
- Two 15,000-gal diesel fuel buried double-wall fiberglass storage tanks
- Mine ventilation heating and cooling surface plant intake at Shaft 1
- Mine ventilation exhaust plant at Shaft 2
- Oil/water separator
- Package extended-aeration sewage treatment plant
- 61,400-yd^3 topsoil stockpile
- 50,300-yd^3 nonsaline mined material stockpile
- 224,100-yd^3 saline material stockpile
- 7.5-million-gallon sedimentation pond with pipe and riser spillway
- 6.5-million-gallon evaporation/retention pond
- 11-acre construction contractor use area
- Site vehicle fuel dispensing facilities
- One 500-gpm and one 250-gpm water supply well
- Fencing
- 1.6-acre offsite parking area.

A brief description of the purpose, functions, and design features of the major support facilities follows.

Potable and fire water at the ESF was to be stored in an above ground welded steel atmospheric tank, with a nominal capacity of 250,000 gal. To provide the required sprinkler system and fire hydrant demand for 2 hours, a dedicated fire water storage of 180,000 gal was provided in the tank. Dedicated space also provided for the ESF daily potable water demand of 34,000 gal. Water supply to the tank was from the ESF water wells, with a combined flow of 750 gpm. Water from the wells was treated ahead of the storage tank by diverting a small flow through the chlorinator system and injecting back into the tank supply pipeline. The water quality for the ESF was to be in accordance with Texas Department of Health and other applicable regulations.

Two 15,000-gal diesel fuel storage tanks of double wall, fiberglass reinforced construction were to be installed underground near the mechanical/electrical building. The quantity stored would allow both 1,200-kW diesel-driven standby generators to operate for up to a 7-day period at full load in the event commercial power supply to the site was lost. The fiberglass storage tanks included an electronic leak detection and monitoring system to continuously monitor both the inner and outer tank walls.

The Shaft 1 supply ventilation system was designed to supply seasonally heated and/or cooled air for circulation through the subsurface facilities. This system included two equal-sized air tempering modules containing (1) direct-fired natural gas air heating furnace, (2) air cleaning section, (3) cooling coil section with a connected condenser section for heat rejection, (4) bypass/mixing section, and (5) supply air fan section. Each of these modules was directly connected to an air supply tunnel, 8 ft wide by 8 ft high in cross section to convey 120,000 cfm of conditioned air to the shaft collar basement. A total 240,000 cfm then entered Shaft 1 through six 6-ft-diameter distribution ports at a slight positive pressure for transport down the shaft.

The Shaft 2 exhaust system provided the complete motive force to circulate air down Shaft 1, through the subsurface facilities, and up Shaft 2 for subsequent return to the atmosphere. This system included six 6-ft-diameter distribution ports in the Shaft 2 collar basement for passage of a total of 240,000 cfm. The air was collected and combined into a single stream in a tunnel 12 ft wide by 10 ft high and conveyed above grade. Immediately after the tunnel surfaces, air was deducted to a "Y" split, with one leg into a single operating vaneaxial fan for return to the atmosphere through a vertical discharge cone. A second equal-sized fan isolated from the system with a guillotine damper was maintained in a fully operational mode for immediate operation should the primary fan malfunction.

The ESF facility provided an oily water separator to remove all free oil from the oily water pumped from the shaft sumps plus that from the mechanical/electrical building floor drain sump. Sanitary wastewater produced during site characterization activities at the ESF was collected by a conventional gravity sewer system and treated at an extended-aeration, package-type treatment plant with a nominal capacity of 0.02 mgd. A lift station was provided to pump the wastewater to the surface plant and also to provide nominal surge capacity for high flow periods such as personnel shift changes. The sewage collection and treatment system would be installed and operated in accordance with Federal and Texas Department of Health regulations.

Areas of the ESF site which were disturbed by construction or grading activities would first have the upper 6 in of soil removed and stored in the topsoil stockpile for restorative use during site decommissioning. The topsoil stockpile would be seeded and fertilized to establish a stand of turf to reduce wind and water erosion and portions of the ESF site disturbed by construction activities and not scheduled for surfacing would also be revegetated.

The nonsaline stockpile was designed to store materials excavated from both shafts from the surface to an approximate depth of 1,000 ft. Excavated materials consisted of clays, silts, sands, and gravels of the Blackwater Draw, Ogallala, Dockum, Santa Rosa, and Dewey Lake Formations (DOE, 1986a). The 50,300-yd^3 capacity provided in the stockpile included a 6-in radial overbreak in the shafts excavation, 20 percent contingency, and 50 percent swell factor for the material. The stockpile would be maintained to assure minimum degradation from erosion and the majority of the stockpiled material would be used to backfill the shafts during decommissioning.

The saline stockpile would store the materials excavated from both shafts below the 1,000-ft depth to the shaft bottoms. Excavated materials were expected to be siltstone, mudstone, sandstone, and anhydrite with salt (DOE, 1986b). Volume of the saline shaft material is calculated the same as for nonsaline material, except a 70 percent swell factor is applied to the broken rock. The saline stockpile was to be constructed on a dual synthetic liner system with leak detection and fluid removal monitoring wells.

The rainfall management system of the ESF consisted of nonsaline and saline systems, each with separate collection and retention facilities. The nonsaline runoff consisted of all areas of the ESF not subject to potential contamination from salt management activities during construction or operations and totaled 46.5 acres. A system of earthen drainage ditches collected surface runoff from those areas, directing the flows to the sedimentation pond. The discharge system allowed only the upper, non-sediment-laden storm water to be discharged. The pond was designed to contain the rainfall volume from a 25-yr, 24-hr rainfall event of 4.96 in, plus 1 ft of freeboard. The operating capacity of the pond was 6.3 million gallons with total capacity with freeboard of 7.5 million gallons.

The areas of the ESF site with the potential to be contaminated by salt management activities during construction and operation totaled 19.07 acres, including the shafts area concrete pavement, site roadway north of the shafts, and the roadways which encircle the saline stockpile and evaporation pond area. A system of lined, trapezoidal drainage ditches collected storm runoff from these areas and conveyed it to the evaporation/retention pond. The pond was designed to contain the rainfall volume from a 500-yr, 24-hr, 7.95-in storm with no losses considered. An additional 2 ft of freeboard capacity was provided. The evaporation pond was to be constructed on a dual synthetic liner system with leak detection and fluid removal monitoring wells. Due to the high evaporation rates and low annual precipitation, disposal of the storm water within the pond was expected to occur through evaporation.

In the western portion of the site, 11 acres was allocated for the construction contractor's use to contain office trailers, stationary and mobile equipment, and construction materials storage.

Water for ESF construction and operations was to be supplied from two wells designed to produce a combined capacity of 750 gpm. Water well WW-1 supplied the construction contractor's water requirements of 250 gpm, while water well WW-2 supplied ESF government-sponsored personnel and ESF facilities water requirements of 500 gpm.

A 7-ft-high chain link fence topped with barbed wire was to be installed around the site perimeter with access through a double cantilever sliding gate or alternate double swing gate which normally would be kept locked. Equipment areas within the ESF include the electrical substation, mechanical/electrical building area, and sewage treatment plant.

A 238-space parking area, based upon the Site Population Study (ONWI, 1987d), was provided outside the ESF perimeter security fence.

6.5 UNDERGROUND TEST FACILITY

6.5.1 Overall Design

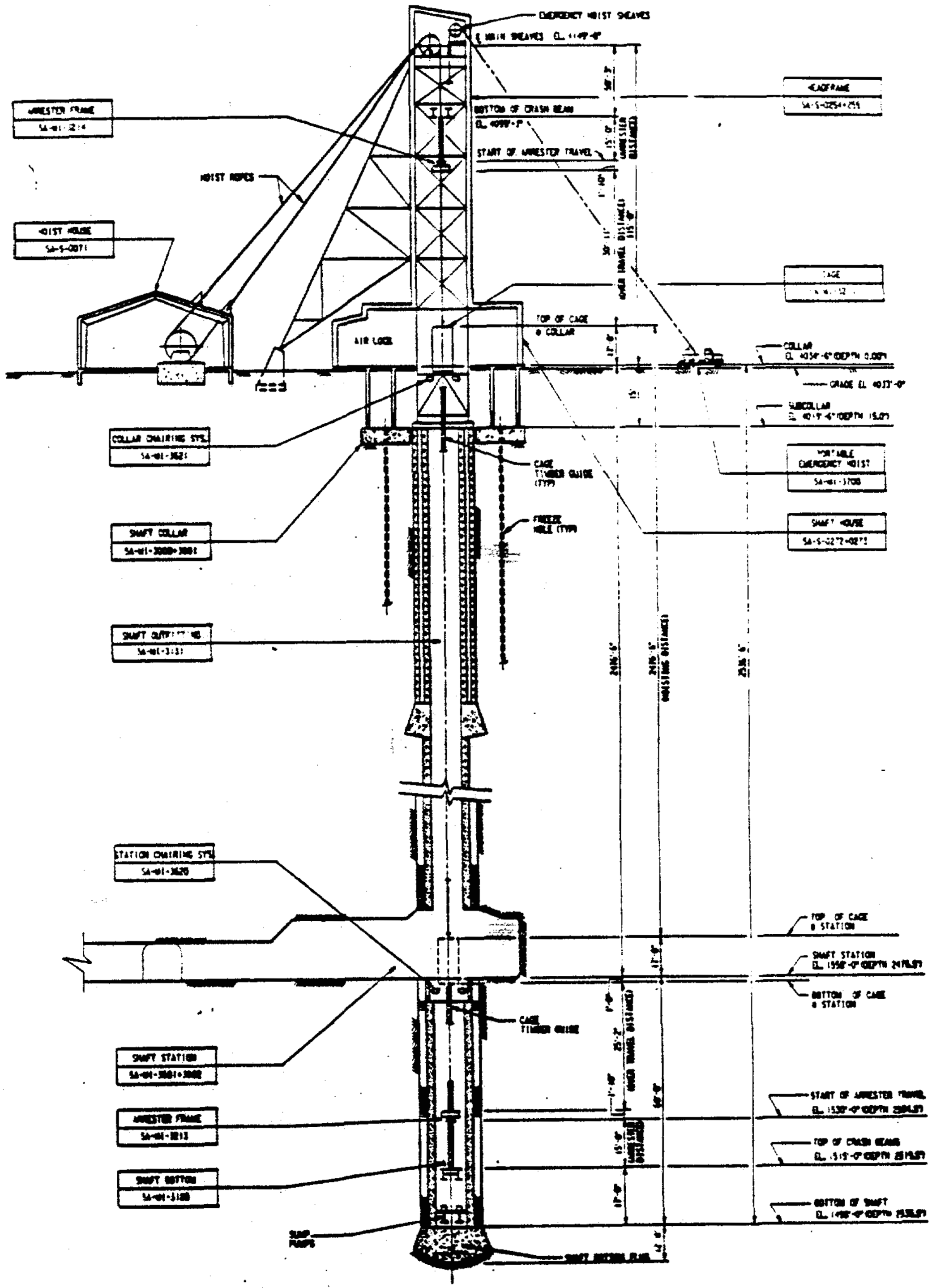
The overall concept of site characterization comprised an underground test facility integrated with two shafts for access and ventilation. Each shaft had a finished dimension of 12 ft in diameter and extended from the surface to approximately 2,547 ft in depth for Shaft 1 and 2,587 ft for Shaft 2. Shaft 1 is shown schematically in Figure 6-2, and Shaft 2 is very similar. The planned construction sequence is shown in Figure 6-3. Initial excavation of each shaft used ground stabilization techniques that included freezing the uppermost 1,000 ft of the Ogallala and Dockum sediments. After the ground in the shaft area was frozen and stabilized, the shaft was sunk to a depth of approximately 1,400 ft using an excavation radius of approximately 10 ft. During this phase, a preliminary liner was emplaced as excavation proceeded down to the 1,137-ft depth. This preliminary liner was composed of either cast-in-place concrete or precast concrete blocks, as dictated by specific ground conditions encountered. The remaining 263 ft would be initially supported by rockbolts, wire mesh, and shotcrete. Following completion of this phase, the final liners would be installed bottom to top from 1,351-ft depth to the collar. The bottom 196-ft final liner consisted of reinforced cast-in-place concrete. The top 1,155 ft incorporated a final liner composed of a steel membrane, concrete, and an asphaltic material in the annulus (between the preliminary and outer steel membrane) to act as a continuous waterproof seal. Upon completion of this upper final liner, the freeze plant was shut off and the ground allowed to thaw. Performance of the shaft liner would continue to be monitored for the life of the ESF.

The remaining depth of each shaft from the 1,137-ft depth to the floor of the shaft station at the test horizon would be excavated to approximately the same diameter. Beyond the shaft station, Shaft 1 would be sunk an approximate additional 75 ft to accommodate a sump and tail shaft area. Shaft 2 would be sunk an approximate additional 120 ft to accommodate a sump, tail shaft, and flexibility area for possible installation of muck loadout facilities for skip hoisting should the facility be expanded at a future date.

During sinking in the lower shaft area, support would be provided in the form of rockbolts and wire mesh immediately following mapping of the newly exposed shaft walls at the shaft bottom. A preliminary cast-in-place liner similar to that installed in the upper section was carried from 1,377 ft to a depth of approximately 2,061 ft, and a foundation installed immediately below that point. Final liner arrangement in this section consisted of concrete, steel, and a sanded cement grout annulus between the steel and preliminary liner. Portions of this shaft increment included compressible material in salt sections only between the shaft rock wall and the preliminary liner to allow for effects of creep in salt strata. The remaining section of the shaft incorporated a liner scheme consisting of shotcrete, wire mesh and rockbolts, compressible resin foam (in salt formations), and cast-in-place concrete.

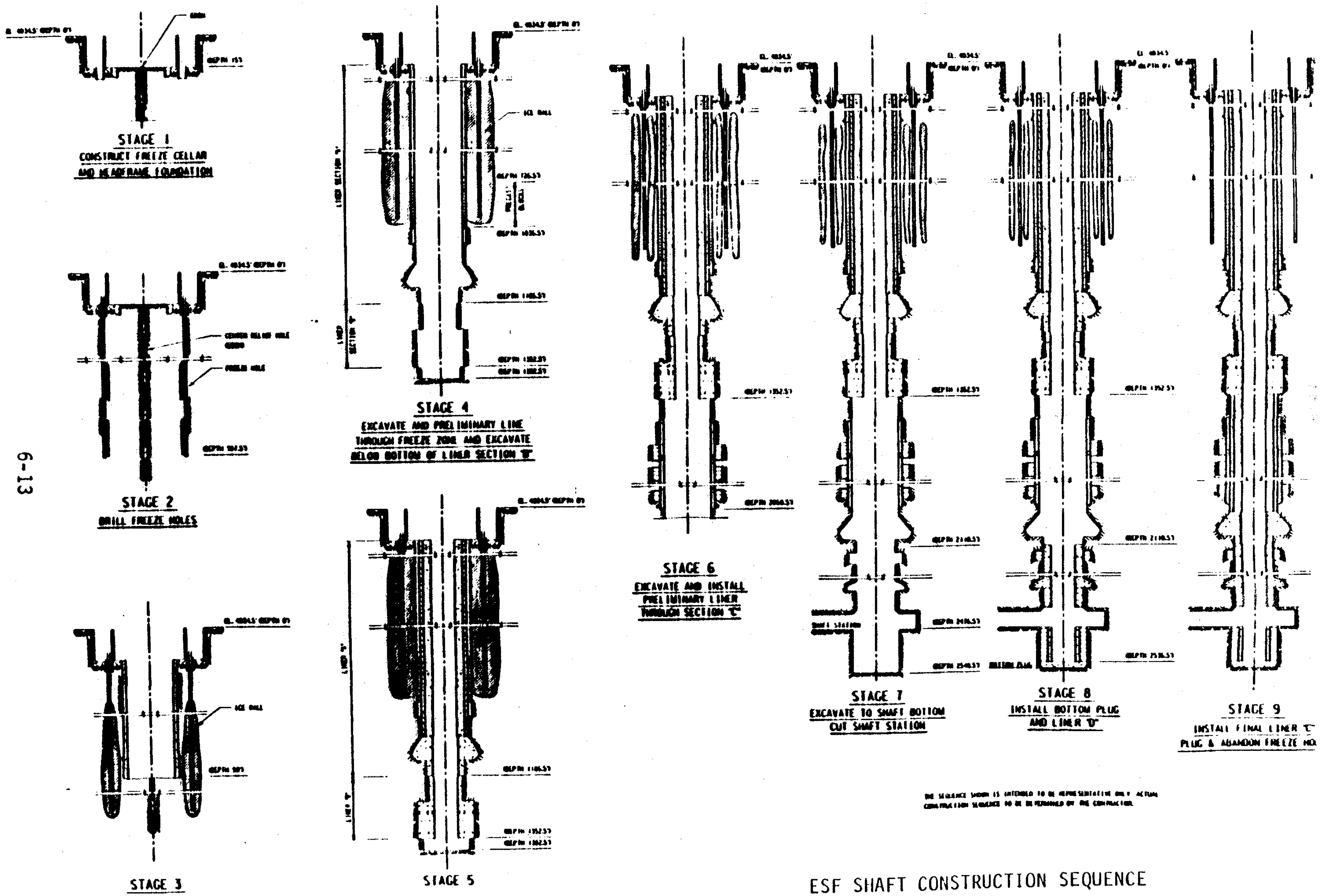
Seals used through the shaft liner scheme were located at the foundation area for the designated final liners. These seals were constructed of various combinations of chemical seal material and cementitious grout, and designed as preclosure operational seals. The design life for shafts, liners, seals, and ground support was 100 years.

After the shafts were completed, the two shaft stations were excavated. Each shaft station was approximately 60 ft long, 21 ft wide, and 18 ft high and provided ample room for assembly of a roadheader and its support systems.



ESF SHAFT CROSS SECTION

Figure 6-2



ESF SHAFT CONSTRUCTION SEQUENCE

Figure 6-3

Immediately after the stations are completed, the shafts were connected by a mechanically excavated drift, which established a ventilation circuit between Shaft 1 (intake) and Shaft 2 (exhaust).

Following establishment of the ventilation circuit, the shafts were outfitted by installing steel sets to carry the conveyance guides and utilities such as compressed air lines, dewatering lines, power supply cables, and instrumentation and communication cables. Shaft 1 accommodated the main service cage operating in balance with a counterweight and served as the main access to the subsurface for personnel and materials. Shaft 2 utilized the sinking bucket arrangement for handling subsurface development material. After the underground development was completed Shaft 2 would be outfitted similarly to Shaft 1, but with two smaller cages operating in balance for limited personnel and material access to the underground level, as well as emergency access.

The subsurface facility design was driven by the following major criteria:

- Provision for the needs of the UTC and Testing Interface Specification (TIS) (Golder, 1987c)
- Provision for overall structural stability
- Provision for a safe environment
- Demonstrate constructibility
- Establish a spatial arrangement that provides flexibility for expansion
- Area must be suitable to support repository construction.

The primary design consideration of the subsurface was to accommodate the UTC in the effort of site characterization using the following major tests:

- Canister-scale heater tests
- Brine migration tests
- Room seal tests
- Drift backfill tests
- Borehole sealing tests
- Mine-by tests.

Except for the initial shaft stations and observation gallery raises, the entire subsurface would be excavated using roadheaders. Mechanical excavation was used to minimize damage to the salt mass surrounding the openings, and to allow instrument installation near to the advancing face with minimum risk of damage. Excavation was anticipated to proceed at a pace consistent with alternate advance and mapping/instrumentation activities.

Underground service systems were designed to support all functions for which the subsurface is designed and included the following:

- Mine heating, ventilation, and air conditioning (HVAC)--main exhaust fans (at Shaft 2) and an air cooling/heating plant (at Shaft 1) for seasonal treatment of intake air

- Communication--emergency telephone system located at strategic positions throughout the subsurface

- Electrical power service--in each shaft and throughout the subsurface as needed

- Compressed air system--in each shaft and throughout the subsurface

- Mine dewatering

- Gas monitoring and alarm system--to sense the presence of methane, carbon monoxide, and other undesirable gases.

Geotechnical monitoring and testing of major design and construction elements in the shaft would include the following activities:

- Geologic mapping
- Ground-water inflow monitoring
- Blast vibration monitoring and seismic velocity survey
- Drilling and core sampling
- Rockbolt testing
- Mechanical and thermal response monitoring
- Absolute stress measurement
- Deformation modulus measurement
- Thermal conductivity and diffusivity testing
- Shaft operational seal testing
- Construction-affected zone testing--decommissioning seals
- In situ methane emission testing.

The subsurface facility also would be instrumented and monitored during construction for the following:

- Geologic mapping and core sampling
- Facility mechanical and thermal response monitoring
- Hydrofracking
- Microseismic
- Ground-water inflow monitoring
- Rockbolt pullout testing
- Gravity surveys
- Electrical surveys.

6.5.2 ESF Shafts

The ESF shafts provided the primary access to the subsurface facility, and must provide safe working conditions in both the construction and operational phases. Extreme importance is placed on the structural integrity of the shafts and their inherent ability to be constructed in a safe and practical manner which minimizes disturbance to the penetrated strata, aquifers, and surrounding rock mass. The basis for the design of the shafts is contained in the Shaft Design Guide (Fluor and Parsons Brinckerhoff/PB-KBB, 1987a) and Input to Seismic Design (ISD) (Fluor and Parsons Brinckerhoff/PB-KBB, 1987b), utilizing data contained in the synthetic geotechnical data base (DOE, 1986b).

Information derived from the synthetic data base indicated difficult ground conditions would be encountered during sinking by virtue of unconsolidated sediments in the Ogallala and Dockum aquifers. Further, it was an absolute requirement that the finished shafts be dry, and not permit commingling of water between aquifers. Approaches to design included analyses of available ground-stabilizing techniques such as freezing, piling, poling, or grouting. Piling and poling are mechanical methods which are simplistic, time-consuming, expensive, and neither watertight nor entirely competent. Grouting forces cementitious material mixed with chemical additives into known water-bearing strata. Because the aquifers in the area are expected to produce large flows of water, it was not possible in the design phase to predetermine the amount of grout needed to achieve ground stabilization.

Ground freezing has been shown to be a cost-effective, time-predictable, and environmentally preferred ground-stabilizing method. Engineering of the freezing process is done by individuals experienced in actual application and proven design techniques. Recent, worldwide experience was brought to bear on individual tailoring of freeze design concepts to the specific ESF geologic data base conditions. A fatal accident during shaft construction in Gorleben, FRG, heightened SRP awareness of the need to critically evaluate the safety of concepts, designs, and the planned construction techniques.

Ground freezing design included computations for parameters such as freeze hole spacing, freeze duration, thermodynamic ground response at differing temperatures, freeze wall radius (inner and outer), and freeze plant capacity. Ground stabilization practice of the Ogallala and Dockum formations by freezing would involve the following operations:

- Drilling freeze holes, temperature control holes, and a shaft center relief hole
- Installing and testing freeze pipes in the ground
- Conducting an ultrasonic survey profile of the unfrozen ground between the freeze holes
- Installing and operating a freeze plant of sufficient capacity
- Circulating a coolant through the freeze pipes
- Conducting an ultrasonic profile of the frozen ground between the freeze holes
- Measuring and evaluating the fluid flow from the shaft center relief hole

- Measuring, monitoring, and evaluating the ice wall creep in the shaft excavation relative to freeze pipe deformation

- Measuring, monitoring, and evaluating the ice wall creep relative to support pressure applied and freestanding shaft wall height

- Measuring, monitoring, and evaluating the thawing of the ice wall relative to freeze hole abandonment schedule requirements and compliance with applicable regulations

- Freeze hole abandonment.

Details on the design for ground freezing are shown in Figure 6-4.

Shaft excavation would be done utilizing conventional drill-and-blast shaft sinking methods, but to achieve minimal wall rock disturbance, the process would be more carefully controlled than is usual for a commercial mine shaft. Methods of controlled blasting and mechanical excavation would be used to minimize rock wall damage, especially in the frozen section where initial rock removal would be done by controlled blasting no closer than a prescribed distance from the final excavation radius.

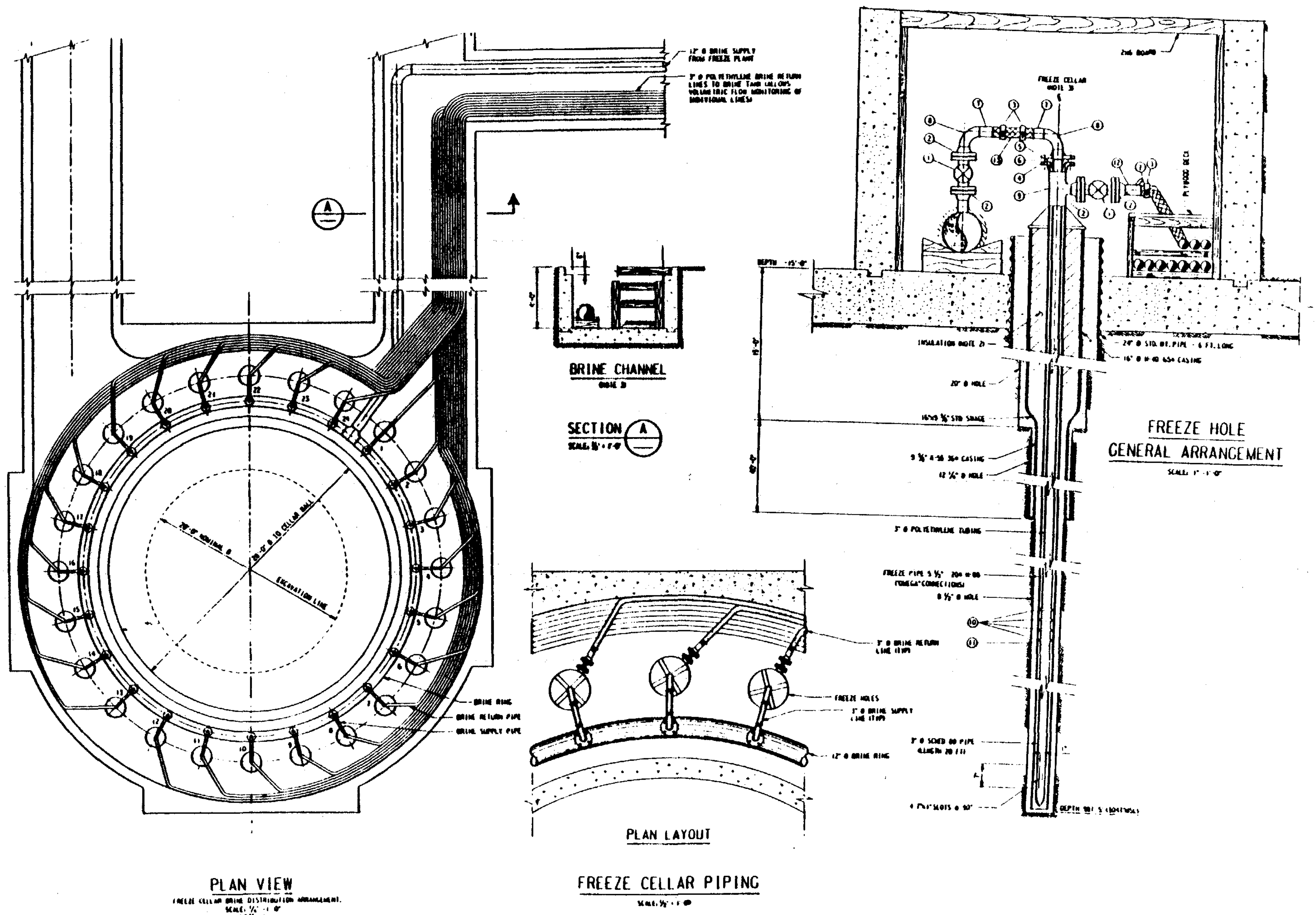
Provision of methods to evacuate personnel from the subsurface was required in case of emergency. Shaft 2 provided this capability to Shaft 1, which was the usual shaft for personnel movement. If both hoists became inoperable, however, an additional emergency egress method was provided using the design shown in Figure 6-5.

In-shaft testing would obtain site characterization-related data, correlate geologic and geotechnical information obtained from the shaft with surface characterization data, and validate ESF design rationale. Because of the diversity of properties and conditions of the strata traversed by the shaft, an extensive in-shaft testing program was identified for both shafts to provide comprehensive validation of the shaft design, especially of the critical design elements. The intent of the project was to fully map both shafts, take photographs, and prepare geologic structures in conformance with Geologic Mapping Procedures (Golder, 1987b) and Shaft Geologic Mapping Criteria (Golder, 1987a). During shaft sinking, samples would also be collected to get information on mineralogy, fluid chemistry and age, and other related information. Information to be obtained from the shafts during excavation is discussed in the Draft Shaft Study Plan for an Exploratory Facility in Salt (Golder, 1987d).

6.5.3 <u>ESF Subsurface</u>

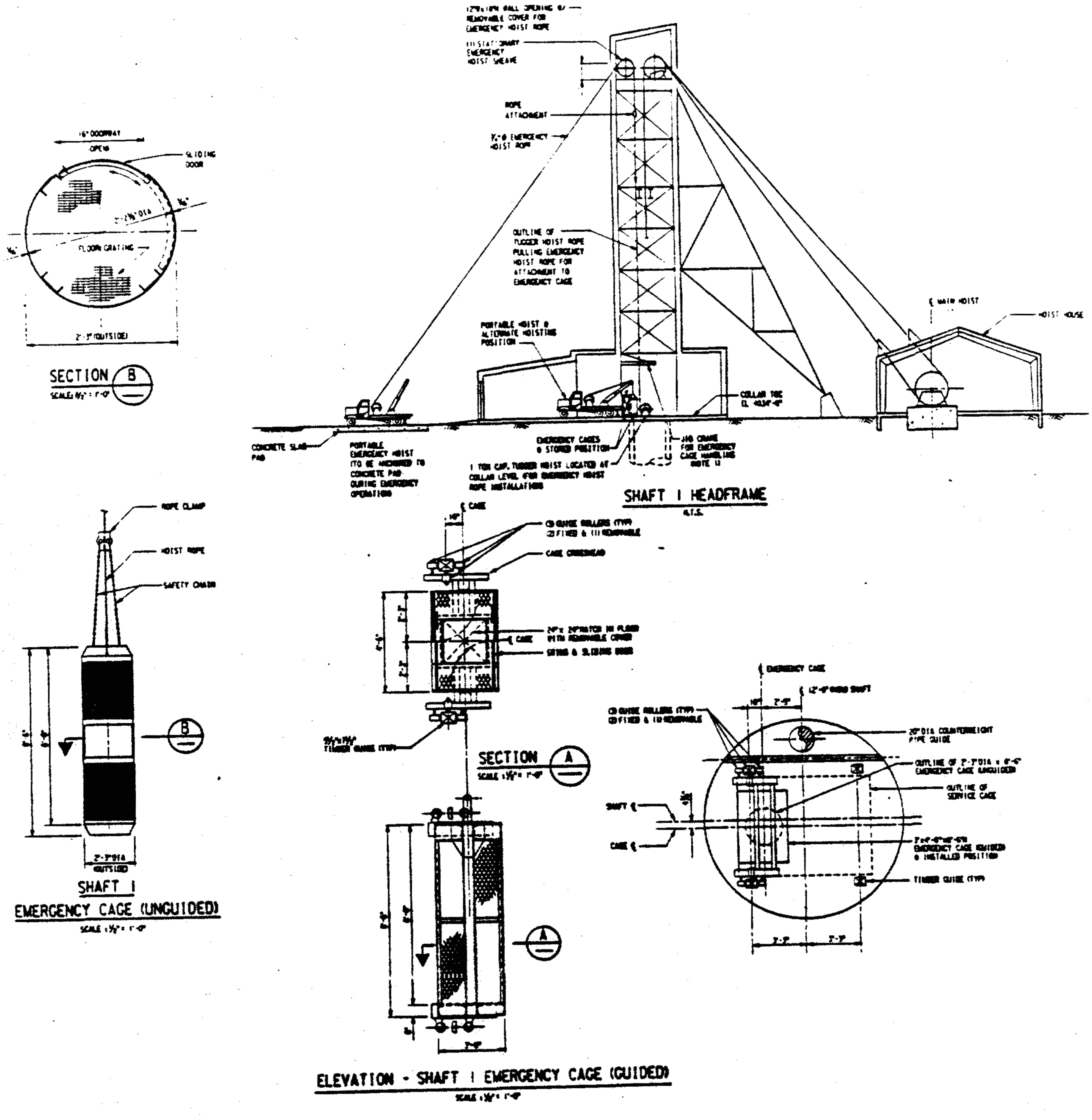
The primary objectives in the design of the subsurface facility were as follows:

- Provide for the needs of the UTC and TIS in site characterization



FREEZE CELLAR AND PIPING DETAILS

Figure 6-4



SHAFT 1
EMERGENCY HOISTING
GENERAL ARRANGEMENT

Figure 6-5

- Demonstrate licensability and constructability
- Provide space and access for maintenance and support activities
- Provide for flexibility
- Avoid design elements which may inherently create preferential pathways promoting ground-water and radionuclide migration
- Provide a safe and efficient working environment from which possible future repository construction may be launched.

The overall requirements for the at-depth facility (ADF) were based on the testing and monitoring requirements to characterize the site. The ADF layout was based on the testing and monitoring, testing support requirements, operational health and safety, ventilation, and other support requirements. In the planning and sequencing of the excavation, consideration was given to the installation of permanent utilities such as power cables, compressed air system, ventilation system, and Local Area Network (LAN) cables for the installation of Automatic Data Acquisition System (ADAS) and data collection units (DCUs). The detailed planning and sequencing provided for online monitoring of the tests as soon as the area to be monitored or tested was available and adequate ventilation was established. The ADC layout provided adequate area to accommodate all of the tests identified, including key/major tests such as mine-by, waste package, room seal test, room-scale heater test, brine migration test, and borehole seal test.

6.5.4 ESF Subsurface Tests

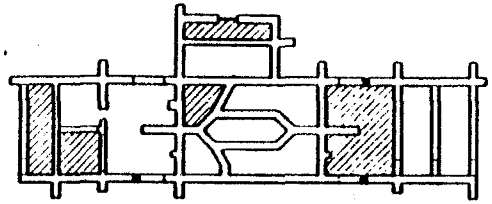
The at-depth facility was designed to meet efficiently the requirements for implementing the at-depth testing program to determine site suitability for development of a repository and to provide information in support of repository design and performance assessment evaluation. The types of tests to be performed in situ in the exploratory shaft are described in the Draft Underground Test Plan for Site Characterization and Testing in an Exploratory Shaft Facility in Salt (Golder, 1987e), and are summarized below.

- Geologic mapping of the ESF including both the shafts, coreholes of various sizes for laboratory testing and for geologic and geochemical testing
- Geophysical logging
- Seismic surveys
- Electrical surveys
- Gravity surveys
- Stress measurement-stress change and in situ stress

- Borehole tracer test
- Borehole convergence and shear test
- Torsional shear test
- Mine-by test
- Rockbolt pullout test
- Mechanical response of the ESF
- Thermal conductivity
- Simulated waste package test
- Room heater test
- Room backfill test
- Thermal response of the ESF
- Single-borehole hydraulic conductivity
- Crosshole hydraulic conductivity
- Tracer diffusion test
- Borehole seal test--in situ
- Room seal test
- Brine migration test
- Grout injection test
- ESF hydrologic monitoring
- Formation fluid sampling--geochemical analysis
- Mine environment testing

The proposed layout of the ESF subsurface and planned test areas are shown in Figure 6-6. Major constraints on the selection of this layout included the provision of separate areas for each of the major tests (for concurrent performance and minimal interference), maximum simulation of proposed repository drift and storage room dimensions and layout, maintenance of fresh air supply to the test areas (especially during construction), and the greatest possible versatility for expansion of the facility according to the flexibility requirements. Drifting of 6,716 ft would be developed parallel and perpendicular to the shaft-connecting airways.

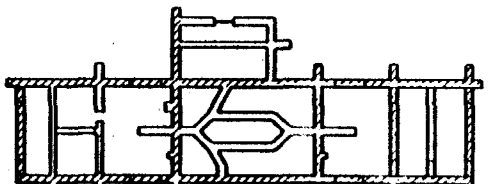
VIEW A
GEOTECHNICAL PERFORMANCE



• CONVERGENCE MONITORING & MPBX'S (TYPE A ARRAYS)

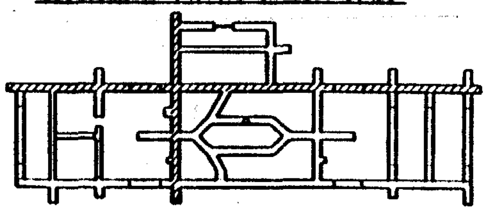
PILLAR CORE CONFINEMENT EVALUATION

VIEW B
LOCATION OF ELECTRICAL SURVEY TRAVERSES AND SEISMIC REFLECTION AND REFRACTION TRAVERSES



D.C. RESISTIVITY AND ELECTROMAGNETIC SURVEY
SEISMIC REFLECTION AND REFRACTION SURVEY

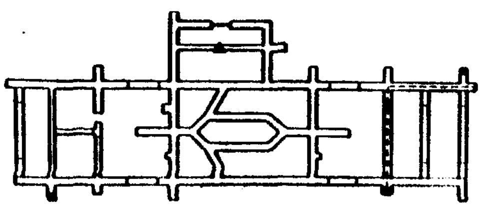
VIEW C
LOCATION OF INITIAL GRAVITY LINES



INITIAL GRAVITY TRAVERSE

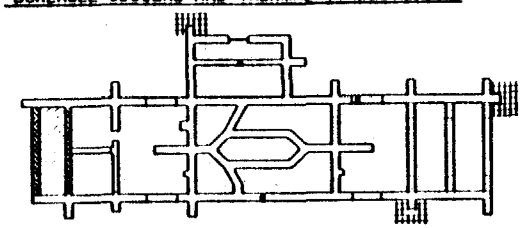
△ UNDERGROUND GRAVITY BASE STATION

VIEW D
ACCELEROGRAPH, ABSOLUTE STRESS, HYDROFRACING & BOREHOLE JACK TESTS



• ACCELEROGRAPH RECORDER LOCATIONS ---- HYDROFRACTURING MEASUREMENTS

VIEW E
BOREHOLE CLOSURE AND THERMAL CONDUCTIVITY



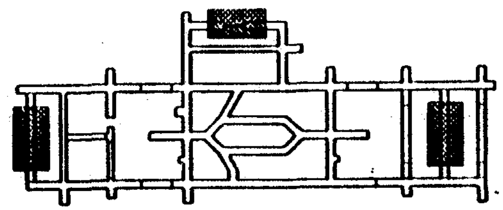
• DEDICATED BOREHOLE CONVERGENCE MONITORING ARRAY

HORIZONTAL THERMAL CONDUCTIVITY PROBE TESTING

VERTICAL THERMAL CONDUCTIVITY PROBE TESTING

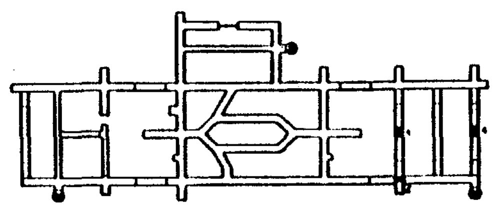
ADDITIONAL HORIZONTAL THERMAL CONDUCTIVITY TESTING

VIEW F
ACOUSTIC EMISSION SENSORS



ACOUSTIC EMISSION SENSOR ARRAY LOCATIONS

VIEW G
CROSSHOLE AND UPHOLE SEISMIC SURVEYS MICRO SEISMIC SURVEYS

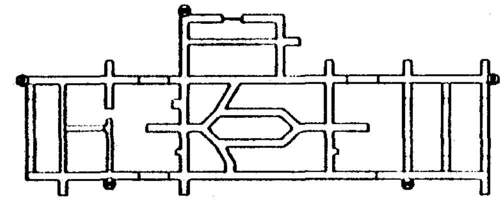


CROSSHOLE AND UPHOLE SEISMIC TESTS

2 NUMBER OF TEST ARRAYS AT LOCATION

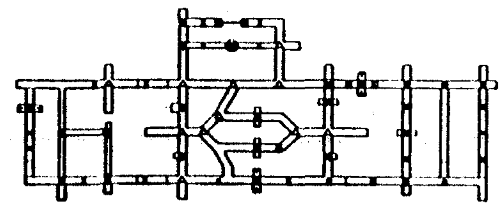
MICROSEISMIC SENSORS

VIEW H
GEOHYDROLOGIC PROPERTIES, HYDROLOGIC RESPONSE AND GEOPHYSICAL BOREHOLE LOGGING



GEOHYDROLOGIC TESTS

VIEW I
FACILITY MECHANICAL AND THERMAL RESPONSE MONITORING



SYMBOL	INSTRUMENTATION	NUMBER OF LOCATIONS
●	TYPE A INSTRUMENT ARRAY	9
x	TYPE B INSTRUMENT ARRAY	21
■	TYPE C INSTRUMENT ARRAY	4
▲	TYPE D INSTRUMENT ARRAY	11
●	TYPE E INSTRUMENT ARRAY	13
O	[illegible] BOREHOLE STRESSMETER ARRAYS	3
◇	HYDRAULIC PRESSURE CELL ARRAYS	3
— —	INCLINOMETER AND TEMPERATURE MEASUREMENT ARRAYS	3
▭	BOREHOLE INSPECTION AND POSSIBLE BOREHOLE SHEAR MONITORING	7

ESF SUBSURFACE INSTRUMENTATION & TEST LOCATIONS

Testing and monitoring of the excavation process itself constituted a significant part of the test program, and these two activities must be carefully integrated down to the level of the excavation cycle itself. All tests not requiring such integration would be located or scheduled after the respective portion of the facility had been completed. Because of the need to excavate the mine-by drift after the placement of the preinstrumentation, the mine-by test was the only large-scale test to be conducted interactively with facility construction.

The excavation sequence of the various drifts would be further constrained by the necessity to prepare sites for, and make operational, the instrumentation trailers. In this way, instruments could be monitored by the Automatic Data Acquisition System (ADAS) as soon as they were installed at or near the face.

6.5.5 ESF Integration Into the Repository

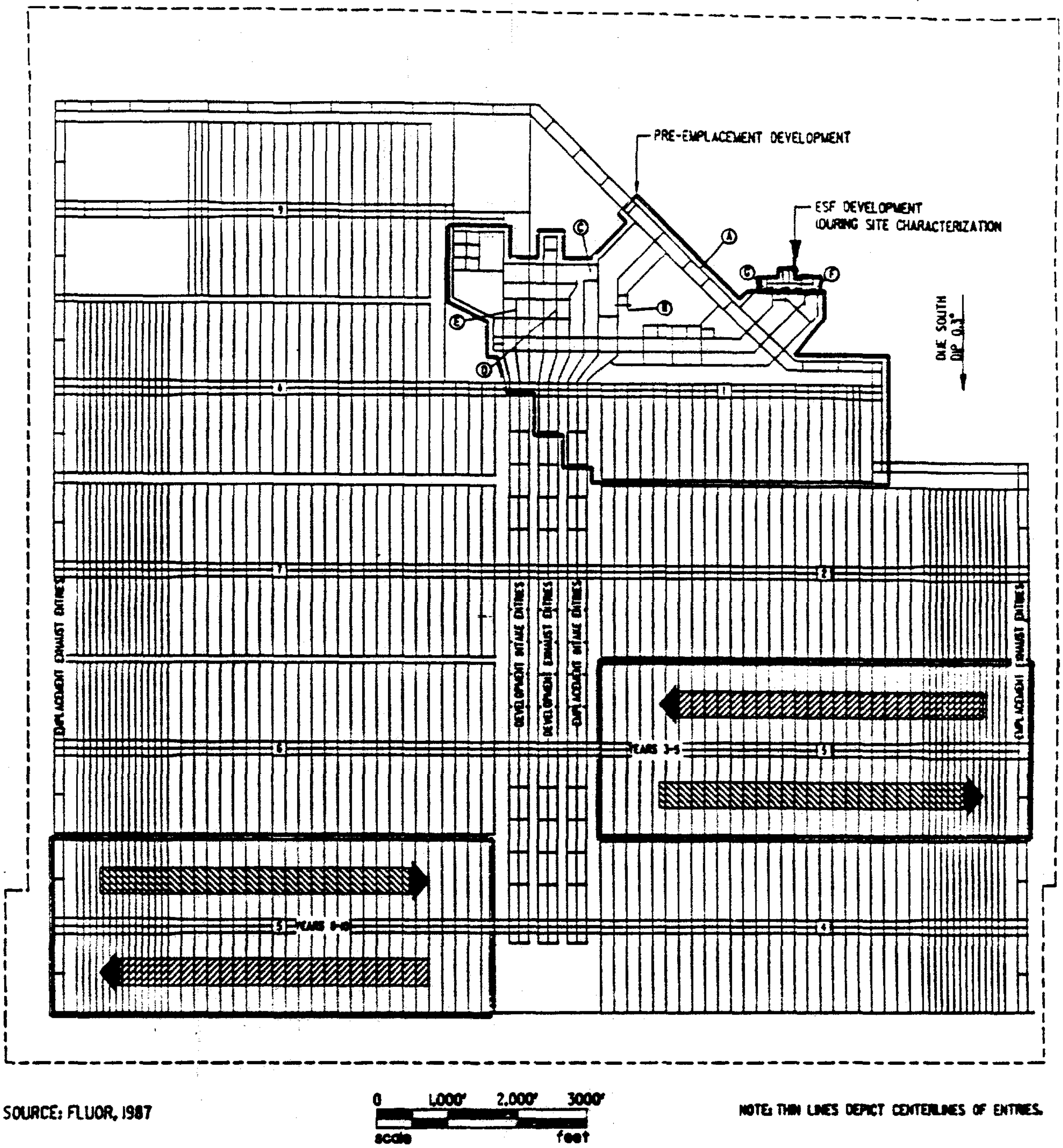
The design of the two exploratory shafts was coordinated with the design of the repository shafts. The two architect-engineering firms responsible for the design of these shafts coordinated their design strategies and designed the five shafts required for the repository and two for the ESF compatibly, based on the same formulas resulting in the same type of preliminary and final linings.

The shaft design coordination was done by the project integrator, Battelle Memorial Institute. Fluor Technology, Inc., architect-engineer for the repository, and Parsons-Brinckerhoff/PB-KBB, architect-engineer for the ESF, developed a shaft design guide which facilitated the design coordination of the shafts and the integration of the exploratory shafts into the repository. The design of all the shafts is in agreement with licensing requirements.

Figure 6-7 depicts the layout of the underground test facility. In the conceptual design of the repository (Fluor, 1987a,b), the two ESF shafts became the two repository emplacement intake shafts. Figure 6-7 shows the resulting location of the underground test facility with respect to the underground area of the repository. As depicted, the underground test facility would be located at the edge of the northeast corner of the repository area. A series of drifts would connect the underground test facility with the main area of the repository.

6.6 FINAL DISPOSITION

If the site were found suitable and selected for the first repository, part or all of the exploratory shaft facility might have been incorporated into the repository design. If the site were not selected for the first repository but considered as an option for the second repository, the site, including all buildings, the headframes, and the access road, would be kept operational and fully secure.



SOURCE: FLUOR, 1987

NOTE: THIN LINES DEPICT CENTERLINES OF ENTRIES.

REPOSITORY UNDERGROUND DEVELOPMENT SEQUENCE

SHOWING LOCATION OF ESF

Figure 6-7

If the site proved unsuitable for further development, it would be restored as closely as possible to its original condition. Site restoration would proceed in accordance with applicable Federal and State regulations, including the following:

- Fill subsurface excavations and the shafts with stored salt and other mined rock, as needed, to provide structural stability
- Limit use of salt or salt-contaminated material to layers below potable water-bearing strata to protect water quality
- Place concrete, polymer seals, clay plugs, or some combination at required intervals to prevent vertical migration of water; one plug would be located at the bottom of the lined portion of each shaft
- Dispose of excess salt off site
- Remove buildings, temporary pipelines, and other surface facilities, including the electricity and communication lines
- Return site to its approximate original contour, including the site access road (topsoil stockpiled on site would be used as needed)
- Topsoil, mulch, reseed, and revegetate the disturbed areas.

Site restoration would be based on the final reclamation plan which would be prepared in accordance with applicable Federal, State, and local regulations. The necessary activities and the sequence could have been as follows:

- Subsurface backfill
- Shaft backfill
- Hoist and headframe removal
- Surface storage area reclamation
- Equipment and building salvage
- Excess salt disposal
- Refuse and site cleanup
- Final grading, topsoil replacement and revegetation.

Plans for site characterization at the Deaf Smith County site excluded the use of high-level radioactive waste. Therefore, no decontamination of the site would have been required after site characterization.

6.7 CHAPTER 6 REFERENCES

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CHAPTER 7

TEST FACILITIES (WBS 7.0)

The natural variabilities of rock masses generally require that, for any significant underground construction project, an underground test facility be constructed in the rock in question prior to embarking on major excavations. In the case of a nuclear waste repository, additional testing to determine the effects of the underground environment on the performance of the facility are required. When the actual site is not available, test facilities may be constructed in similar geologies to provide preliminary data. This chapter discusses the test facilities planned and operated for the salt repository project.

7.1 BACKGROUND

It was recognized from the work performed at Project Salt Vault (Bradshaw and McClain, 1971) near Lyons, Kansas, that additional underground testing was necessary to answer many of the questions that were being raised relative to construction, operation, and performance of a nuclear waste repository in salt. BPMD entered into negotiations with International Salt Corporation to lease space in the Avery Island Mine, an operating salt mine near Lafayette, Louisiana, to perform a series of tests which will be discussed in Section 7.2.1. Also, the United States Government entered into a bilateral agreement with the Federal Republic of Germany to cooperate in a test program to be performed in the Asse Mine near Hanover, Germany.

In addition to the implementation of the field testing mentioned above, plans for testing to be conducted in an underground test facility to be constructed at the candidate repository site as part of site characterization have been developed.

7.2 FIELD TESTS

7.2.1 Avery Island Mine, Louisiana

Field experiments were conducted by the ONWI in a test facility at the 168-m (550-ft) level of the Avery Island Mine in domal salt between 1978 and 1984 (Van Sambeek, 1980; Van Sambeek et al., 1981; Blankenship and Stickney, 1983). The mine is located near New Iberia, Louisiana. The objectives of this testing program were

- To evaluate the response of domal salt to a heat source typical of a HLW package and to observe corrosion rates on potential waste package materials (i.e., Waste Canister Heater tests)

- To examine movement of natural and synthetic brine in a temperature field and develop methods for future fluid-migration experiments (i.e., Brine Migration tests)

- To measure salt material constitutive properties under controlled boundary conditions (i.e., Heated Corejack tests)

- To evaluate radar techniques for detecting structural discontinuities and other anomalies within intact salt and for locating waste canisters

- To determine the in situ permeability of heated and unheated salt

- To evaluate borehole closure under stress.

The results from these field investigations and supporting studies are as follows:

1. Good agreement ($\pm$ 10 percent) was obtained between the predicted and measured thermal response for the heater tests, with generally satisfactory agreement for displacements, although some difficulties were related to extensometer anchor slippage. Vibrating wire stressmeters and strain gages generally failed to function.

2. The thermomechanical response of the domal salt was generally similar to that of bedded salt, with no unanticipated behavior.

3. Corrosion rates were greatest (approximately 1 mm/year) for carbon steel, with minimal attack on Zircaloy-2 and type 304L stainless steel.

4. The rates of brine movement under thermal gradient were generally within the range predicted by the simplified brine migration models, although there are several possible mechanisms of migration and the relative contribution of each is not well understood.

5. The Heated Corejack test results did not substantiate using the currently applied exponential-time creep formulation for Avery Island salt. Longer duration laboratory creep tests using different load paths are required to establish a reliable empirical creep law and associated parameters, particularly at lower temperatures.

6. Radar scanning should prove useful for evaluating certain geologic structures in salt formations typical of Avery Island salt and for relocating emplaced materials during repository operations.

7. In situ salt permeability is lower than that determined from laboratory samples, and the permeability appears to decrease with increasing temperature.

8. The borehole closure tool was not sufficiently sensitive for the small closures observed at Avery Island, and the experiment suffered from reliability problems associated with positioning of the borehole closure tool.

7.2.2 Asse Mine, Federal Republic of Germany

The Asse Mine, located east of Hanover, West Germany, is the research and development facility for underground testing for the Federal Republic of Germany (FRG) waste disposal program. The facility is an abandoned salt and potash mine situated in an anticlinal structure.

The United States and the FRG have a bilateral agreement to exchange waste management information and have initiated joint field testing in the Asse facility (Westinghouse, 1983a; Rothfuchs et al., 1984). The primary objectives of the Asse Brine Migration tests, which commenced in 1983 at a depth of 750 m (2,470 feet) within the Asse test facility, are

- To observe thermally induced fluid migration under different boundary conditions (e.g., borehole pressures), with and without radiation
- To qualify testing methods and equipment to be used in brine migration testing
- To observe conditions in the borehole resulting from the arrival of migrating water, including radiolysis, corrosion, gas generation pressure, and other synergistic effects
- To observe thermal and mechanical behavior of the salt in the presence of heat, brine, and radiation, and the effects of these on the salt mechanical properties
- To obtain data on room closure.

This testing is now completed (Rothfuchs et al., 1987) and the results indicate

- Good agreement between predicted and measured temperature fields
- A smaller amount of water released to the pressurized waste package boreholes than to the unpressurized holes, indicating that gas pressure has some influence on the fluid migration mechanism
- Insignificant differences between tests incorporating radiation and similar tests not incorporating radiation, indicating that radiation has little influence on the fluid migration mechanism
- Calibration difficulties with strain gage stressmeters.

7.3 UNDERGROUND TEST PLAN

An Underground Test Plan (UTP) (Golder Associates, 1986) was prepared in which are described the characterization and testing activities to have been performed within an Exploratory Shaft Facility at the Deaf Smith County, Texas, salt site. The Texas salt candidate repository site was to have been characterized in detail to provide information necessary for development as a repository, for repository design, and for performance assessments required as part of the licensing process. Such detailed site characterization would have included surface-based site investigations and testing, laboratory studies, and tests and investigations performed within an Exploratory Shaft Facility.

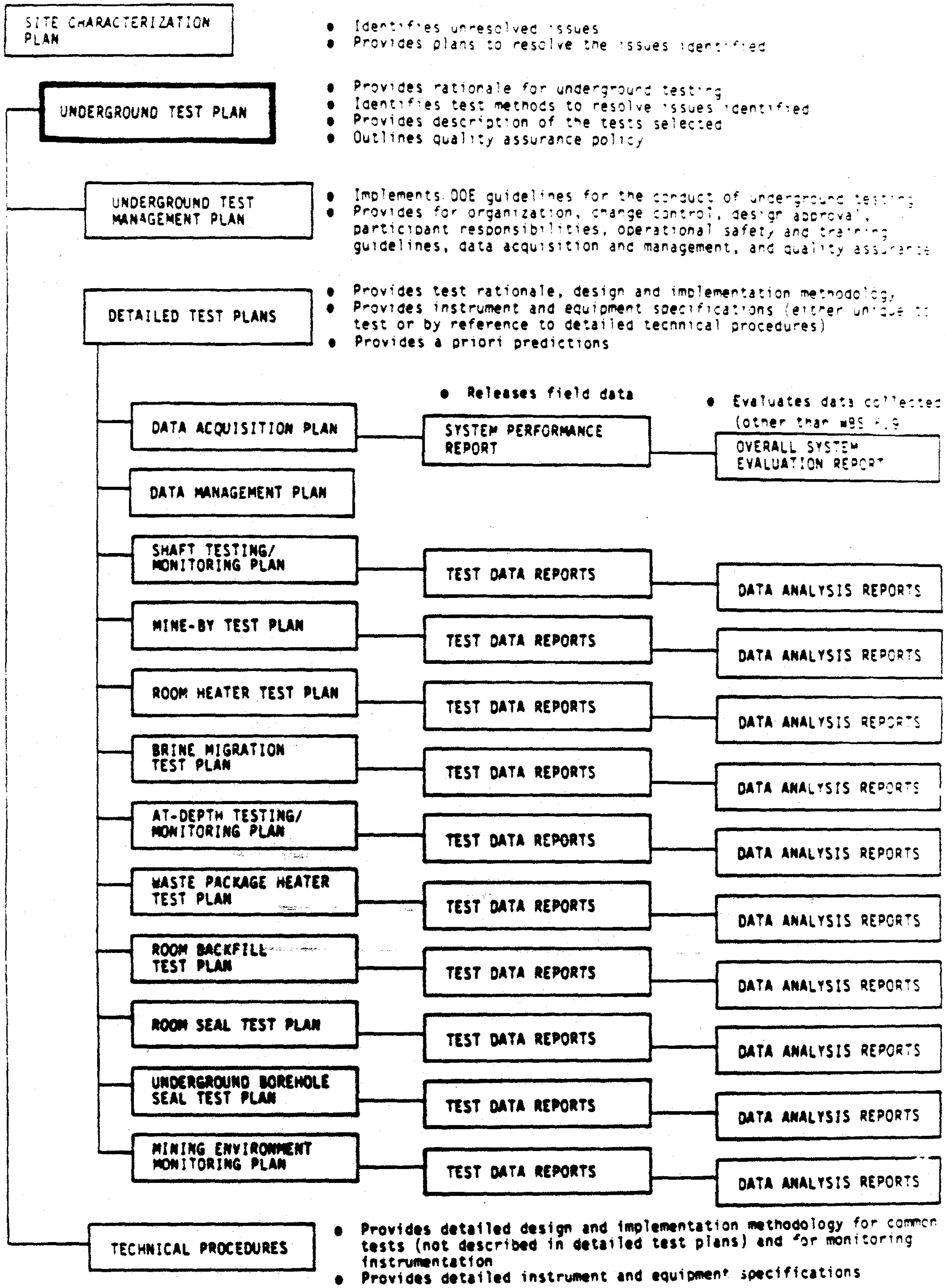
The UTP addresses one part of the detailed salt site characterization effort. The scope of the Plan included the development of the information needs to be addressed by underground testing and the corresponding conceptual level test program. Subsequently, detailed test plans and associated technical procedures were to be developed for each of the tests. The relationship of the document to the overall hierarchy of documents associated with underground testing at an ESF in salt is shown in Figure 7-1.

Future updates of this Plan will reflect any changes that may better define the information needs to be satisfied by underground testing at the Deaf Smith County site. These changes may result from modifications to the reference disposal concepts, additional site specific information from ongoing site characterization activities, further definition of the licensing strategy (i.e., identification of those components/processes of the repository system for which credit will be taken in performance evaluations), and quantitative sensitivity studies to refine the information needs to be addressed by underground testing.

The UTP is consistent with the Nuclear Regulatory Commission's "Generic Technical Position on In Situ Testing during Site Characterization for High-level Nuclear Waste Repositories," as well as with all applicable regulations, orders, and policies.

The UTP is organized according to the following:

- A brief summary of related salt testing experience.
- Brief descriptions of the Deaf Smith County site, and reference repository and waste package designs.
- The rationale for the test program. This rationale consists of the development of information needs starting with the regulatory and design requirements, and the identification of the recommended underground tests based on consideration of all available surface-based, surface borehole, laboratory, and underground test methods. Supporting information is presented in appendices.
- The selected individual tests are presented, together with a description of the proposed ESF and a discussion of the schedule for performing the test program. Conceptual level test descriptions are given in an appendix.
- The Quality Assurance requirements.
- A list of references and a glossary at the end of the main text.



HIERARCHY OF DOCUMENTS FOR UNDERGROUND TESTING

Figure 7-1

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CHAPTER 8

LAND ACQUISITION (WBS 8.0)

The objective of the Land Acquisition task was to organize, plan, schedule, budget, monitor, control, coordinate, and report all land acquisition work for the Salt Repository Project (SRP), including strategy and plans and their execution for land access and protection, cooperative agreements, and rights and easements. This included all efforts for the acquisition of licenses, leases, fee simple titles, withdrawal agreements, cooperative agreements, and other agreements which indicate an interest in surface and subsurface lands for principal boreholes, other geotechnical and environmental activities, exploratory shafts, or repository.

Stone & Webster Engineering Corporation, the SRP Geologic Program Manager for Texas, retained a land agent who obtained land for some monitoring boreholes that were drilled in Texas off the Deaf Smith County site. The work done so far in regard to land acquisition at the Deaf Smith County site is summarized in the draft Land Acquisition Plan (LAP) (SRPO, 1987).

8.1 LAND ACQUISITION PROCESS

SRP did not acquire any land at the Deaf Smith County site. The progress made toward the acquisition is described in the LAP, the content of which is highlighted below. The reader is referred to the LAP for details.

The policy for obtaining the property interests to be acquired by the U.S. Department of Energy (DOE) to conduct site characterization and to protect the nine-square-mile site was to obtain only the minimum property interests essential to achieve DOE's mission. The dominant Federal policy under the Uniform Relocation Assistance and Real Property Acquisition Policies Act was to make every reasonable effort to acquire expeditiously real property by negotiation. The DOE supported this policy, and consequently, obtaining negotiated agreements was the Department's goal in carrying out the land acquisition program.

The U.S. Army Corps of Engineers (COE) was DOE's land agent for the acquisition process. The COE prepared a Real Estate Planning Report which recommended to DOE the nature of each interest DOE needed for land protection and for completing the site characterization and site investigations at the Deaf Smith County site. The COE Real Estate Planning Report is included in the LAP as an appendix.

8.2 SRP LAND ACQUISITION PLAN

The LAP describes the land acquisition activities necessary to conduct site characterization and the related site investigations in Deaf Smith County, Texas, and to protect the Deaf Smith County site from those human activities which could adversely affect the licensability of the site as a repository.

This plan provides information on the following:

- Land acquisition for site characterization and site investigations
    - Geotechnical
    - Environmental/socioeconomic
    - Security
    - Land protection
- Legal requirements
- Budgetary considerations
- Policies, options, selected land interests, and mitigation measures connected with site characterization and land acquisition
- Implementation of the plan.

8.3 CHAPTER 8 REFERENCES

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CHAPTER 9

PROJECT MANAGEMENT (WBS 9.0)

9.1 PROJECT CONTROL

The objective of the Project Control task was to plan, implement, direct, and administer all activities required to effectively control the cost, schedule, technical quality, and scope of the Salt Repository Project (SRP).

The broad scope of the Project Control task assured that all aspects of the SRP effort were focused to support the successful completion of the project's mission within the constraints of the currently operable DOE waste management policies and strategy.

In 1979, as the Salt Repository Project grew and increased in complexity, the need for a formalized system of program and project management became critical. It was estimated that at least 85 percent of the total project funds would be expended either under ONWI subcontract or by other DOE contractors for which ONWI would have overall technical coordination responsibility and funds accountability. Effective communication of project progress was imperative among the ONWI Program Manager, his department managers, the subcontractors, DOE, and their contractors. To ensure that project cost, schedule, and technical performance was effectively accomplished, ONWI established a formal Project Control System to support their contract work and to assist in the integration of the total SRP activities. The three types of information entering the ONWI Project Control System were cost information, schedule information, and technical scope.

The Work Breakdown Structure (WBS) elements were the basis for project responsibility assignments, planning and scheduling, resource allocation and budgeting, work authorization, procurement management, technical performance, interface control, and for contract reporting. WBS element descriptions were created to the detail necessary to ensure full understanding of the scope of work so that schedules, budgets, and technical performance requirements could be aligned with the descriptions.

The project control requirements, as set forth in the contract, the statement-of-work, specifications, and associated documentation, constituted the basis for development of a cost/schedule plan. All project effort was authorized and controlled by project management through formal authorization documents. Functional management directed and monitored the activities of their organizations in compliance with formal agreements that had been arrived at between project management and themselves, as reflected in the authorization documents. Figure 9-1 shows an example of an early WBS and Figure 9-2 shows the latest SRP WBS.

In October 1983, the Cost/Schedule Control Systems Criteria (C/SCSC) of DOE Order 2250.1 was applied by ONWI to its existing management system. The Battelle Project Management Division's Project Management System (PMS), as applied to the Salt Repository Project, received a validation as part of the CH10140 contract.

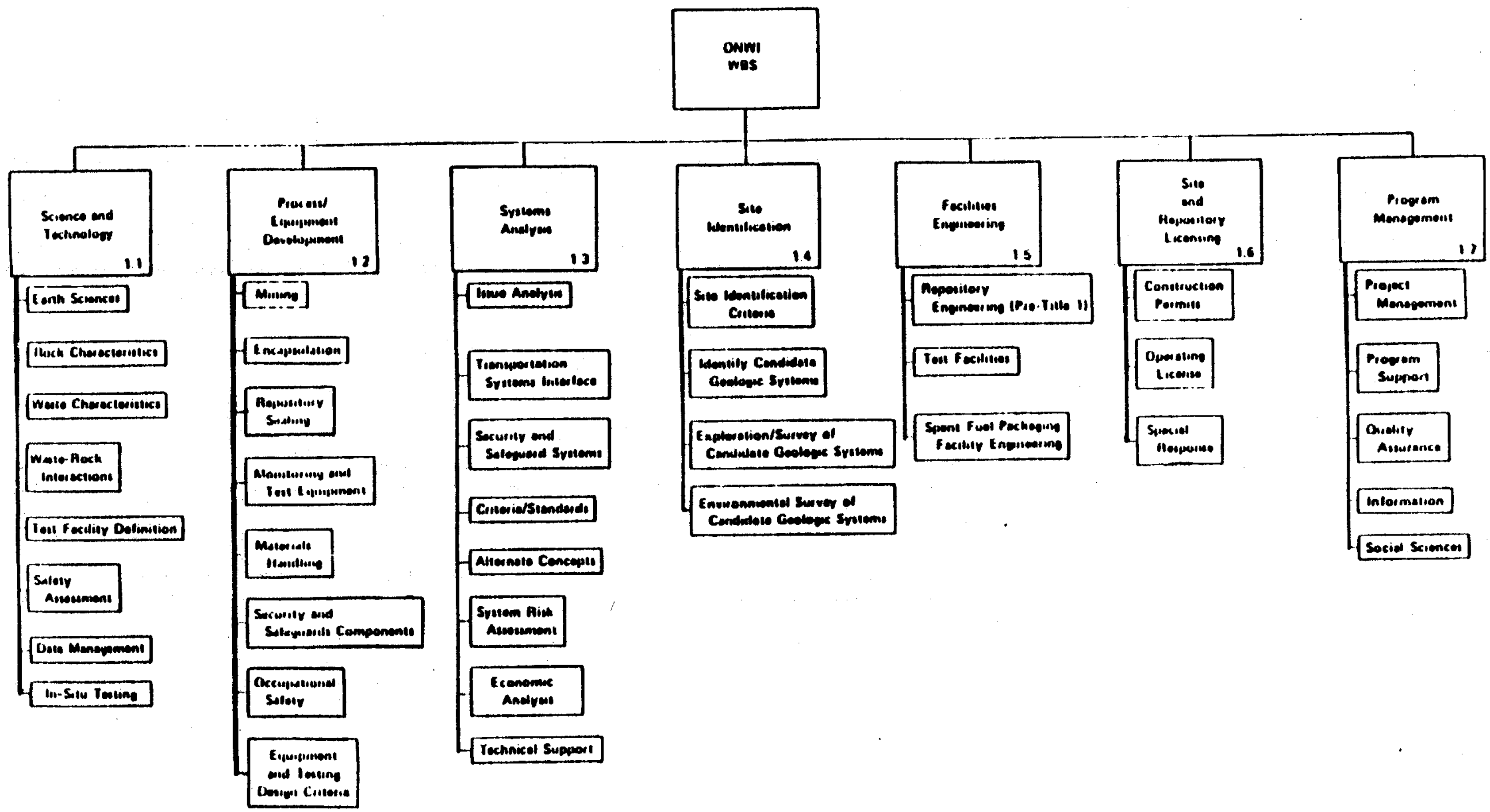

Figure 9-1. Early ONWI Work Breakdown Structure

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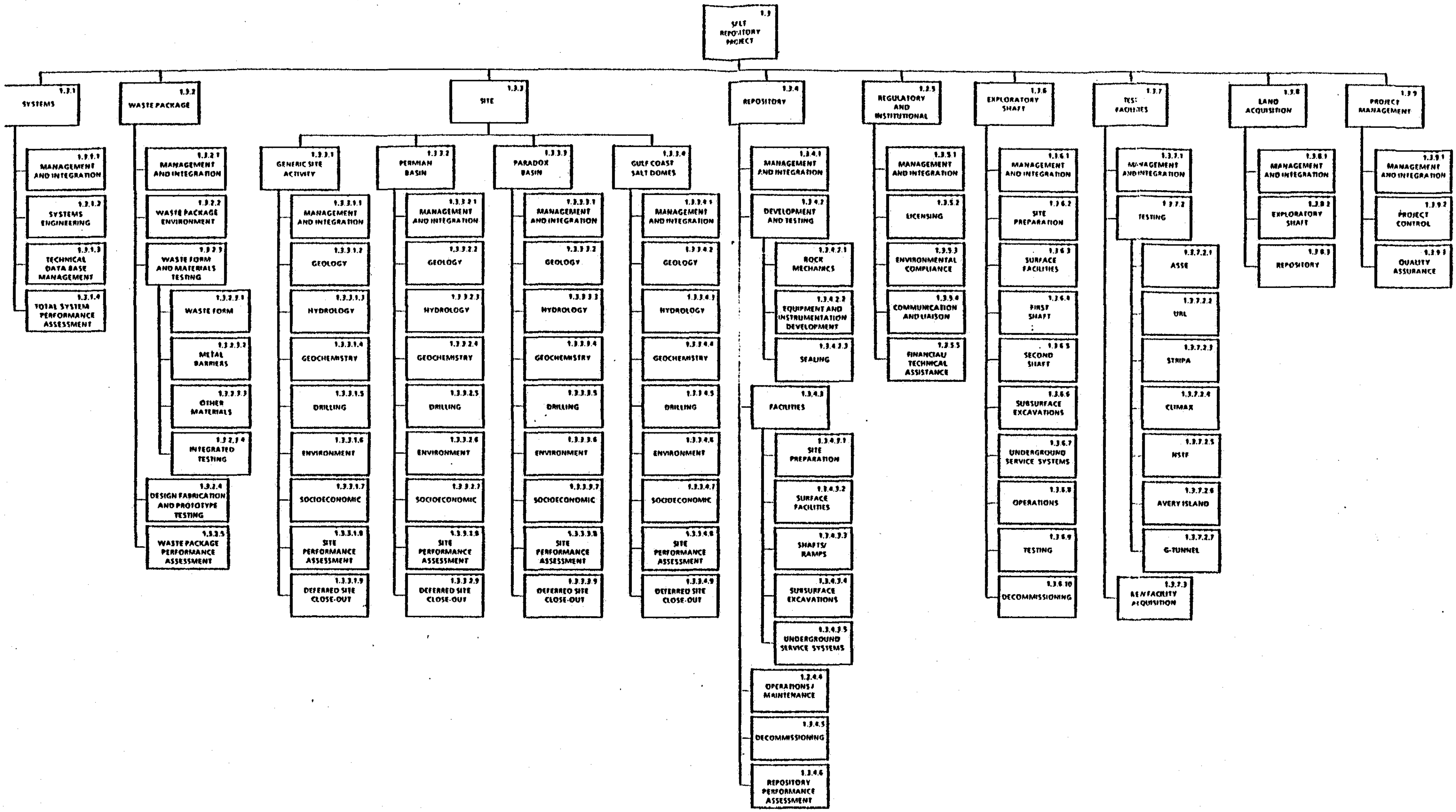


Figure 9-2. Contract Work Breakdown Structure

The project control methodology for the Salt Repository Project consists of (1) developing an integrated project plan utilizing the PMS, (2) controlling and revising the integrated project plan, and (3) assigning PMS managerial responsibilities within the organizational units responsible for cost and schedule management. This involved integrating, organizing, planning, and controlling the work.

9.1.1 Integrated Planning

The DOE requirements set forth for this project, including the contract statement of work and associated documentation, constituted the basis for developing the integrated project plan. All Salt Repository Project work was authorized and controlled by DOE project management through formal authorization documents. ONWI's management directed and monitored activities of its own organization in compliance with formal agreements made with DOE's management, as reflected in the authorization documents.

The project control approach began with a planning effort that translated DOE guidance into specific contract goals and requirements. In addition to annual and periodic guidance from DOE, the status of current activities could also generate a need for replanning.

9.1.2 Organizing Work

Work was organized by developing the Contract Work Breakdown Structure (CWBS), cost accounts, and work packages. The approved CWBS and its related dictionary was used by a contractor as the single common structure for defining project work. The CWBS was end-objective oriented and included the activities necessary to effectively address the DOE-established Project Summary WBS (PSWBS) objectives. In addition, the CWBS defined the hierarchical relationships among elements of planned work.

A functional organization structure reflected the manner in which a contractor organized its staff to accomplish the defined work. Integration of functional organizations with CWBS elements was necessary when assigning functional responsibility for the work to be performed. A Responsibility Assignment Matrix was used to communicate this integration. The intersection of functional organizations and CWBS elements resulted in the designation of cost accounts, a key management control point in the contractor's system. Cost accounts were further subdivided into planned and authorized work packages, which defined in detail the scope, schedule, and resource requirements for accomplishment at the work package level. Planning was summarized into cost accounts and then summarized by PSWBS elements.

9.1.3 Planning Work

Within the CWBS, and based on summary logic networking for the strategy of project objectives, a Milestone Schedule (MS) was finalized depicting major project objectives and related events to be accomplished at specific dates.

With the MS and the CWBS established, project interrelationships and durations were more fully developed through the networking technique. This technique identified and defined major events or milestones to complete the project and established the necessary constraints through logic diagramming (networking) to establish a step-by-step plan for accomplishing all work under the WBS elements. The major activities of the SRP, along with their durations and interrelationships, were developed and published as the Salt Repository Project Networks (SRPO, 1987).

Estimated time and resource requirements were established for each WBS element and ultimately for the entire project through this process. The entire process was iterative and involved all major project participants, including DOE prime contractors, integrated contractors, and subcontractors. Figure 9-3 shows the establishment of project planning elements when the integrated project plan is initially completed.

As the integrated project plan evolved beyond the WBS summary level, organizational elements required to accomplish the project work were identified. As appropriate to the planning process, roles and missions of ONWI and other Salt Repository Project components were incorporated as they were defined. Once the integrated project plan was defined and networked, resource requirements were developed at the work package level and summarized through the cost accounts to the WBS and organizational elements at all levels and for the total project. When resources and schedules were balanced and the integrated project plan was established, budgets were finalized for all ONWI organizational and WBS element levels. These budgets, developed at the cost account and work package levels, became firm management commitments at all organization levels within the project. All project effort was thereby totally defined, properly integrated into the whole, and assigned to specified individuals at the work package level for accomplishment within planned resources and schedules.

The fully evolved planning process provided the ingredients necessary to authorize, assess the status of, and redirect the entire project throughout its life cycle. For most fiscal years a project plan was prepared (ONWI, 1982). Once the project plan was established, replanning was accomplished through a formal process that assured use of one (and only one) integrated project plan at all times. Status assessment and reporting were accomplished in conjunction with established plans for both ONWI management and DOE overview of project status. Figure 9-4 depicts the formal planning process.

9.1.4 Controlling Work

After initial planning was completed, the project was controlled through the formal program control process. Networks, schedules, and cost account plans were the basis for this control procedure. During the planning process, all work was defined to the level of planned work packages. Prior to initiating the effort on these work packages, cost account managers confirmed cost, schedule, and work scope requirements for each package, and an authorized work package was processed to begin the work.

If the work package was to be released precisely as originally planned, work package authorization proceeded promptly. If any changes were required in

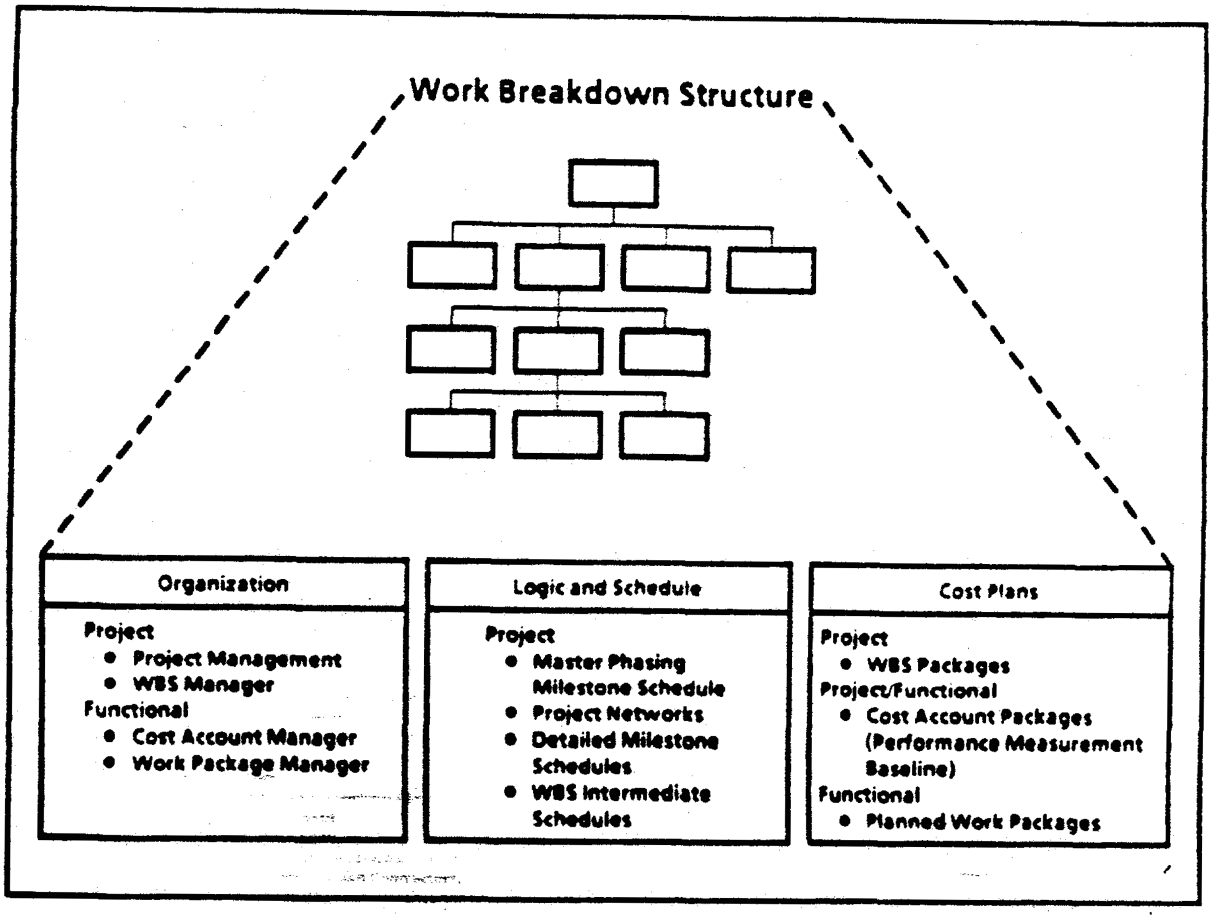


Figure 9-3. Establishment of the Integrated Project Plan

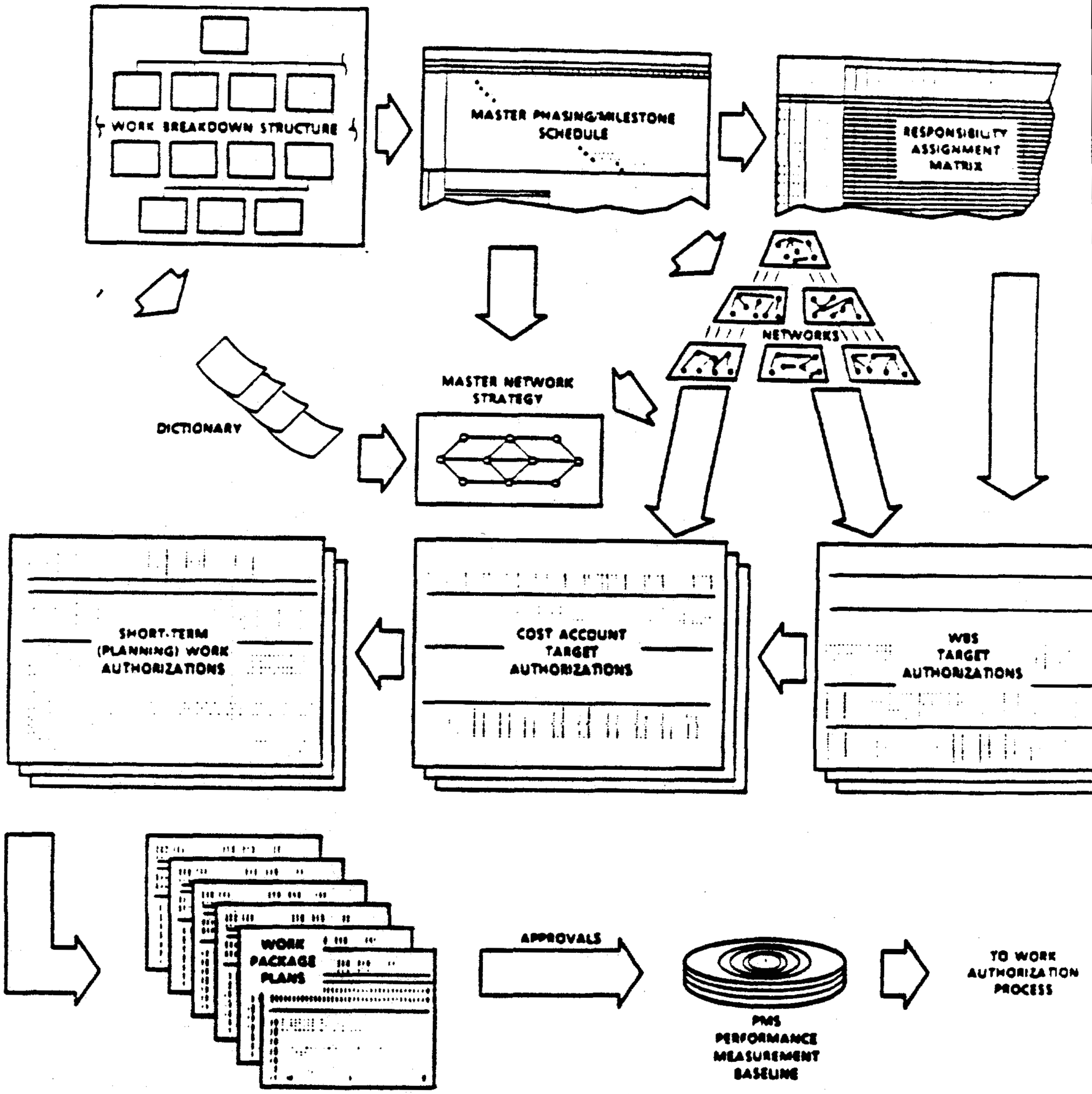


Figure 9-4. Planning Process

scope, schedule, or cost, a Package Change Record (PCR) was prepared to realign all elements of the plan.

Project status and resulting forecasts to complete remaining work (latest revised estimate) were determined via the earned-value techniques, which were an integral part of the formal validated PMS. This technique translated project status into data elements that compared actual cost of work accomplished with work planned.

The data elements shown on Table 9-1 were displayed for internal reports and for client reports from the same data base. On-line access to these data elements was provided to SRPO and DOE Headquarters. When established variance thresholds were exceeded, management explanations were petitioned and corrective actions were initiated.

In addition to earned-value reporting, which expresses schedule status in terms of dollar variances from the plan, calendar milestone status was regularly determined and reported via the computerized schedule subsystem module. This technique used the planning networks as a basis for expressing status in terms of days and weeks ahead or behind schedule. The technique produced critical path data and forecast dates based upon current reported milestone status.

Information was provided to both project and functional management on a regular basis, usually monthly, thus supplying current cost/schedule and the calendar schedule status in summary for the total project, by WBS element, and by functional organization at any desired level. Figure 9-5 depicts the program control process throughout the project life.

9.1.4.1 Cost Control

All cost planning, budgeting, control, and reporting were accomplished in accordance with formal procedures that are a part of the validated PMS and the accounting system published procedures. These procedures and ongoing practices resulted in computerized output reports that showed budgets at all levels, reported actual costs versus budgeted costs, and reported variances relative to those budgets.

The previously mentioned reports presented information in both WBS and organizational formats and were distributed to program and functional managers in the appropriate formats. Variances for both cost and schedule were contained in the same reports as an integrated data set. Summary levels as required by the contract were reported to SRPO. Data for internal contractor reports and for external client reports were the same at given levels and originated in the same data base.

9.1.4.2 Schedule Control

Schedule updating, analysis, control, and reporting provided visibility into the entire project effort. Both networks and bar charts were used to report schedule status; however, the different methods emphasized different aspects of reported conditions. Bar chart schedule status depicted progress

Table 9-1. PMS Data Elements

Data Element	Description
BCWS	Budgeted cost for work scheduled. This is the time-phased resource plan based upon the work scope as scheduled for each applicable work package. BCWS was then summarized to a cost account and then to a WBS element.
ACWP	Actual cost of work performed. This is the actual cost reported at the performing organization's work package level which was then summed to a cost account and then to a WBS element.
BCWP	Budgeted cost for work performed. This is the planned value earned for work actually completed through the current reporting period. The BCWP included the budgeted dollars earned for all discrete and level-of-effort work accomplished through the reporting period.
SV	Schedule variance. This is the difference between work performed (BCWP) and work scheduled (BCWS).
CV	Cost variance. This is the difference between work performed (BCWP) and cost recorded in the Category (ACWP).
FTC	Forecast to complete. This is the time-phased forecast of costs for authorized work remaining.
LRE	Latest revised estimate. This combines the cumulative-to-date ACWP and the current FTC for the remaining work.
BAC	Budget at completion. This is the total planned cost for a specified scope of work.
VAC	Variance at completion. This value is the difference between BAC and LRE.

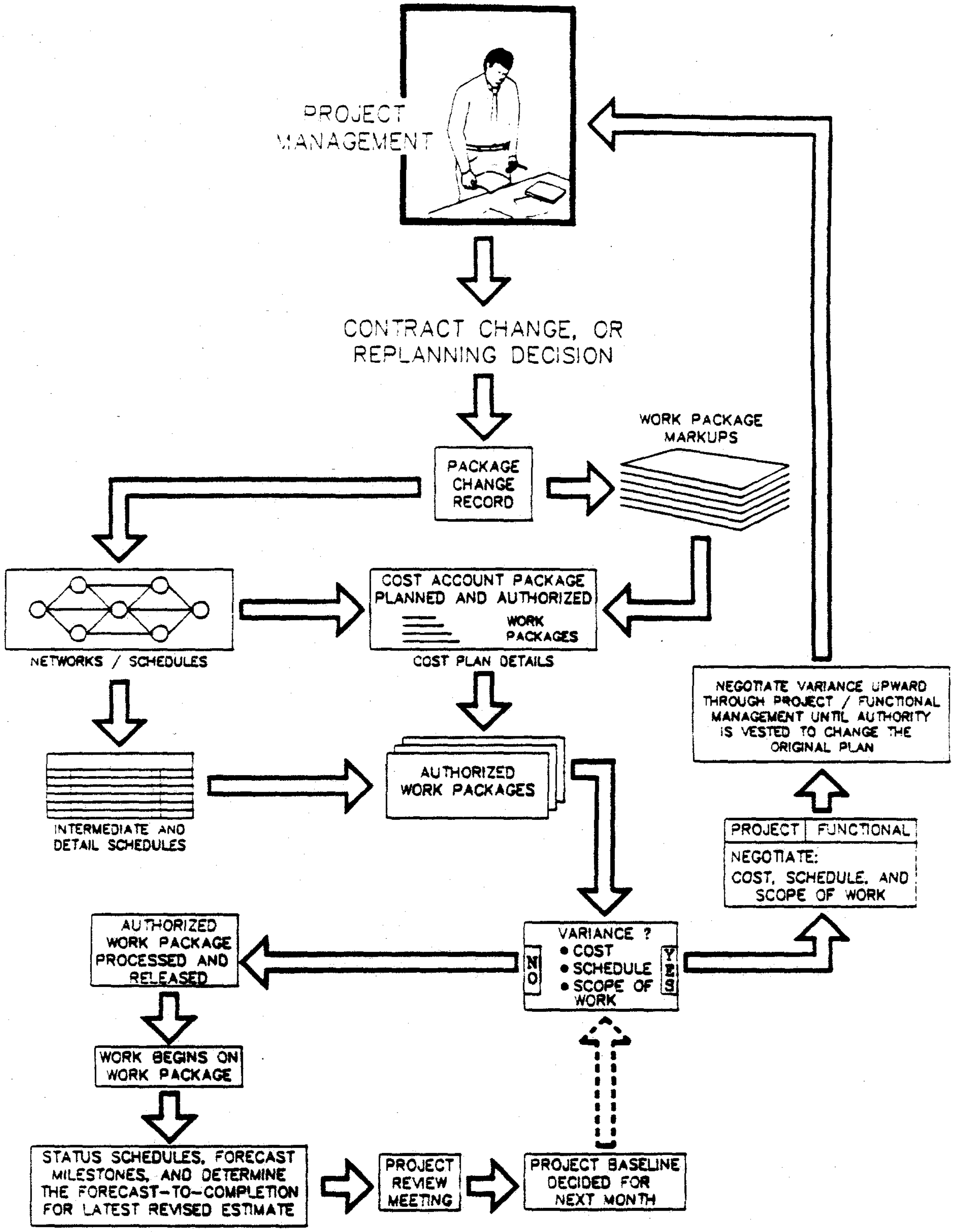


Figure 9-5. Program Control Process

compared with the near-term (30 to 90 days) schedules, requiring considerable analysis to highlight impacts on subsequent events. Network processing depicted the impacts of near-term status on deliverables and project milestones and was based on forecasts contained in the scheduling subsystem data base.

At the end of each month, all milestones due in that month (or due in prior months, but not completed) and all milestones due for completion within the next 90 days were updated by the responsible work package manager to reflect the current status. If the milestone had been completed in accordance with the documented criteria, actual dates were noted. If the milestone had not been completed, a forecasted completion date was provided. Milestones designated as objective indicators were utilized for earned-value computation. Networks available from the network system were as follows:

- Master (consistent with the Milestone Schedule)
- Summary levels (by Level II of the WBS)
- WBS (detail operating networks)
- Combination (special requests for mixed WBS levels or special functional schedules).

By utilizing the detail operating networks, project critical paths were identified and schedule problem analysis reports (SPAR) were generated. The SPAR focused on near-term major milestones (DOE, Major Systems Acquisition) and addressed corrective action to be taken, including any logic changes that could be made to reduce negative slack or accomplish events as scheduled.

During the monthly project reviews, key milestones identified in the SPAR that had negative slack in excess of one month were discussed.

9.1.4.3 Variance Analysis

Cost and schedule variances to baseline plans were analyzed each month when variance thresholds were exceeded. The cost subsystem used detailed work package data for performance measurements, for projection of estimates, and for analysis and actions at the cost account and WBS levels. The schedule subsystem used work package objective indicators and other milestone events in networks that provided an interrelationship between deliverables and activities. Functional management and project management were notified of variances to plans.

In the first week of each month, milestone completions, or forecasts of completions, were provided at the work package level. Schedule conditions causing more than a month's criticality to the project were analyzed for impact on future milestones. In the second week, external costs were incorporated by accounting. In the third week, cost/schedule performance reports were given to managers. Variances exceeding tolerance thresholds were highlighted for analysis and corrective action.

A monthly program management meeting was convened to review variance conditions, to analyze potential problems, and to investigate alternative plans that could eliminate those variances. Functional management reported on cost accounts at variance. Project management directed corrective actions or recommended changes in project work requirements.

9.1.4.4 Status Reporting (MSSR, CPR, PSR)

The SRP project status reporting was summarized primarily in the following deliverable data:

- The status of major Salt Repository Project milestones was provided in the Milestones Schedule and Status Report (MSSR)
- The cost/schedule variance explanations were provided in the Cost Performance Reports (CPR)
- These reports, together with an overall assessment of the ONWI work, were contained in the Project Status Report (PSR) and submitted to DOE/SRPO by the twentieth calendar day of each month following the period being reported. A consolidated PSR for the total Salt Repository Project was submitted by the twenty-eighth calendar day.

Status reporting related project cost, schedule, and technical accomplishment to a baseline plan, within the framework of both the WBS and the organizational structure.

PMS information was generated from the PMS data base which summarized and displayed cost and schedule information at the end of the previous month. This information was used by the department and project managers in reviewing the status of their cost accounts/work packages in preparation for the monthly management review and in writing their narratives and explanations for the monthly status reports. All reporting was accomplished in accordance with formal PMS procedures. These procedures resulted in computerized output reports that presented data in both WBS and organizational formats. The reports contained BCWS, BCWP, and ACWP. Variances for both cost and schedule were also contained in the same reports. Summary level information as required by the contract was reported to DOE in compliance with the checklist form (DOE F1332.1) reporting requirements.

9.1.4.5 Reviews

Monthly management reviews were a key element in the contractor internal management process. The reviews (one for contractor management and staff, the other for SRPO) provided the contractor Program Manager, the functional and department managers, and other staff with a comprehensive yet concise overview of the current project status, as well as a more detailed summary of major accomplishments. A primary objective of each review was to anticipate problems before they occurred, to assess the possible effects of potential problems on all elements of the project, and to formulate effective corrective action. Each review meeting focused on financial status and cost variances, procurement status, milestone status, detailed project status to the reporting level of the

WBS, and priority efforts for the next month. Prior to the management reviews, the accomplishments, potential problem areas, and corrective actions were discussed in the narrative section of the monthly PSR prepared for DOE.

Semiannual project reviews were scheduled by DOE. These reviews involved representatives from DOE-HQ, SRPO, and contractor management, as well as other interested agencies. The reviews followed a format similar to that for monthly management reviews, but emphasis was placed on the areas of highest significance, on longer term trends, on the relationship of the current project to activities projected for future years, and on a comprehensive evaluation of current and future years' budget status and Salt Repository Project strategy.

9.1.4.6 Revisions

Replanning the Performance Measurement Baseline resulted from a redirection (or reschedule) of authorized and planned work. Replanning may have included the scope, milestones, or time-phased resources, and sometimes may have included all of them. When these actions affected in-scope effort, they were called "revisions." When there was new work or different work due to the SRPO direction, these actions were called "changes." In either case, revisions and changes were visible and traceable through the PMS baseline maintenance procedures.

All replanning was formally controlled and signed off. Retroactive replanning was prohibited. Revisions were traced through the Program Management Reserve and changes were tracked through the Undistributed Budget. Appropriate entries were logged into the Budget Base Transactions Log and records were kept monthly for historical evidence of the baseline replanning.

The Battelle Performance Analysis and Control (BPAC) display system provided on-line access to the PMS data. The BPAC system allowed SRPO and contractors to view information at the WBS, cost account, and work package levels. BPAC graphics and tabular reports assisted the user in the project control process.

Because all project effort was planned at the work package level, all replanning was documented on separately numbered Package Change Records (PCRs). Any replanning must have at least been signed by the parties (organizations) to the original agreements, and other contractor managers must have signed any replanning that affected their part(s) of the project.

9.1.5 Budget

DOE Order 5700.7B was issued regarding the Work Authorization System (WAS). This Order established a formal process for budget development, authorization, and monitoring of the client's funded work performed at all the specified contractor facilities.

An annual budget proposal was prepared in agreement with guidance from the DOE as reference case for the budget submission to Congress. The SRP prepared annual budgets, taking the form of an FY budget submission. With various scopes and parameters, alternative budget cases were also developed and recommended. In addition, budget exercises were developed for "what if" project strategies.

The Field Work Proposal (FWP) served management needs primarily by providing a vehicle for SRPO approval of a contractor's work statement at the highest feasible budgetary level consistent with maintaining program control. An FWP consisted of an identifiable group of associated tasks or activities. It was measurable in terms of performance and included specifications of verifiable events or deliverables marking project achievement. A Budget Planning (BUDPLAN) Data Base was used to process the WAS data, and to assure the reconciliation between the WAS budget (when approved) and the cost/schedule baseline data.

9.1.6 Summary

The following Project Control objectives were accomplished:

- Developed cost, schedule, scope of work, and budget/funds requirements
- Incorporated requirements into an integrated cost and schedule baseline
- Reported performance against the cost/schedule baseline
- Provided technical, cost, and schedule information and analysis
- Developed and maintained cost plan, manpower plans, and performance reports
- Formulated strategic guidance necessary to perform successful project planning and to resolve SRP issues
- Developed standard approaches to defining project goals consistent with the overall network
- Monitored progress toward established milestones and recommended corrective action in the event of deviations
- Integrated all functional elements to ensure proper technical input to meet established milestones by ensuring, through the planning process, that all necessary inputs were properly scheduled and resources allocated
- Supported SRPO in the preparation and review of Project Management Information for all SRP participants
- Coordinated the Annual 5-Year Budget Submissions to ensure all items were consistent with project needs
- Prepared inputs for Section 8.5 of the Site Characterization Plan, and coordinated the Section 8.3 schedule content.

9.2 QUALITY ASSURANCE

The DOE was required to maintain an effective quality assurance program for all work affecting quality performed in support of the SRP. The program was planned, documented, implemented, and maintained to support activities as the SRP progressed from site investigation and characterization toward final design. The DOE required that the SRPO and all of its SRP participant contractor organizations also implement quality assurance programs appropriate to their activities affecting quality. These quality assurance programs were developed in accordance with federally mandated requirements, which included implementation of self-assessment activities to ensure compliance with these requirements.

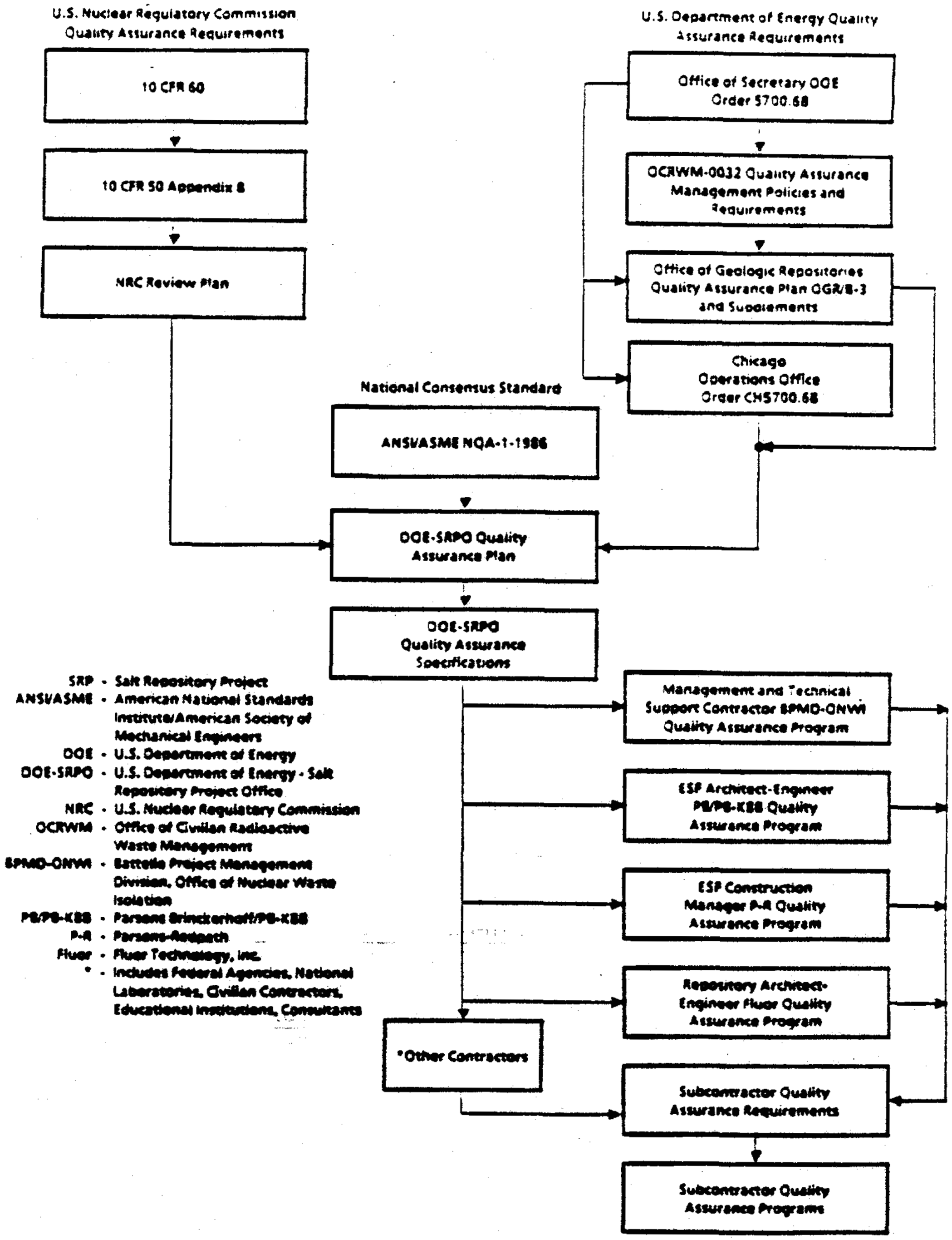
The goals of the quality assurance programs of the SRPO and its participant contractor organizations were to ensure that research, development, demonstrations, scientific investigations, and subsequent construction and production activities were performed in a controlled manner, and that resulting technological data were valid and retrievable. The quality assurance requirements and measures necessary to be included in the quality assurance programs were consistent with the items' and/or activities' "importance to safety" and "importance to waste isolation," and the achievement of DOE mission objectives.

9.2.1 Quality Assurance Program Requirements

The important early activities during site exploration of host salt rock primarily concentrated on accumulating data from published information and the performance of geologic surveys. The determination of specific locations involved more complete investigations that included drilling of boreholes and laboratory testing/analysis that focused on the engineering and isolation characteristics of the host rock. The emphasis of quality during these activities was placed on techniques and procedures developed within the scientific community where their reliability had been demonstrated by broad usage. Subsequently, the emphasis toward quality assurance during these activities addressed more programmatic requirements which included performance of technical reviews, verification of work performed, records management, and independent third-party verification of contractor performance by the SRPO and SRP participant contractors.

Quality assurance programs of the SRPO and SRP participant contractors continually evolved from 1981 to the present through the incorporation of new or revised requirements imposed by government regulations.

The quality assurance requirements for the SRP originated from two primary sources: the U.S. Nuclear Regulatory Commission (NRC) and the U.S. Department of Energy (DOE) (see Figure 9-6).



Quality Assurance Documents Hierarchy

Figure 9-6

The NRC, through Title 10, Code of Federal Regulations (CFR) Part 60, Subpart G made the quality provisions of 10 CFR Part 50, Appendix B mandatory for all systems, structures, components, and activities designated as "important to safety or important to waste isolation. Additionally, the NRC published the NRC Review Plan (1984), which provided the criteria and methods the NRC proposed to use to review the quality assurance programs for site investigation and characterization during the prelicensing phase, and to provide guidance for establishing an acceptable program for items and activities designated important to safety and important to waste isolation.

The DOE, prior to the publication of 10 CFR Part 60, issued DOE Order 5700.6A (1981) to provide policy, set forth principles, and assign responsibilities for establishing, implementing, and maintaining plans and actions to ensure quality achievement in DOE programs.

The directive further identified that in nuclear-related programs, the American Nuclear Standards Institute/American Society of Mechanical Engineers (ANSI/ASME) document NQA-1 was the preferred standard for quality assurance (ASME, 1986). Although ANSI/ASME NQA-1 provides direction for quality assurance programs during the design and construction phases of nuclear facilities, the DOE determined that this document provided an adequate basis for interpreting the quality assurance requirements of 10 CFR Part 50, Appendix B as applied to nuclear waste repositories, without contradicting its requirements or guidance.

Consistent with DOE Orders 5700.6A and subsequently 5700.6B (1986), and to ensure uniform and acceptable interpretation of the requirements for quality assurance for nuclear waste repositories, the DOE Office of Geologic Repositories (OGR) issued its "Quality Assurance Plan for High-Level Waste Repositories," OGR/B-3 (DOE, 1984; revised 1986). This plan set forth program-wide quality assurance requirements and defined the OGR and the project management responsibilities for quality assurance. The requirements also applied to DOE field offices and contractors who were assigned responsibilities for performing and verifying activities affecting quality. The plan also referenced the use of ANSI/ASME NQA-1 as the standard for providing the detail needed to meet the requirements of 10 CFR Part 50, Appendix B, and the NRC Review Plan.

Subsequent to the issue of the revised OGR Quality Assurance Plan, the DOE Office of Civilian Radioactive Waste Management issued <u>Quality Assurance Management Policies and Requirements</u> (QAMPR) (DOE, 1985b), which set forth overall integrated quality assurance management policies and requirements and provided a general basis for the development of more detailed quality assurance management policies and requirements. The OGR Quality Assurance Plan incorporated, by subsequent revision, reference to the QAMPR.

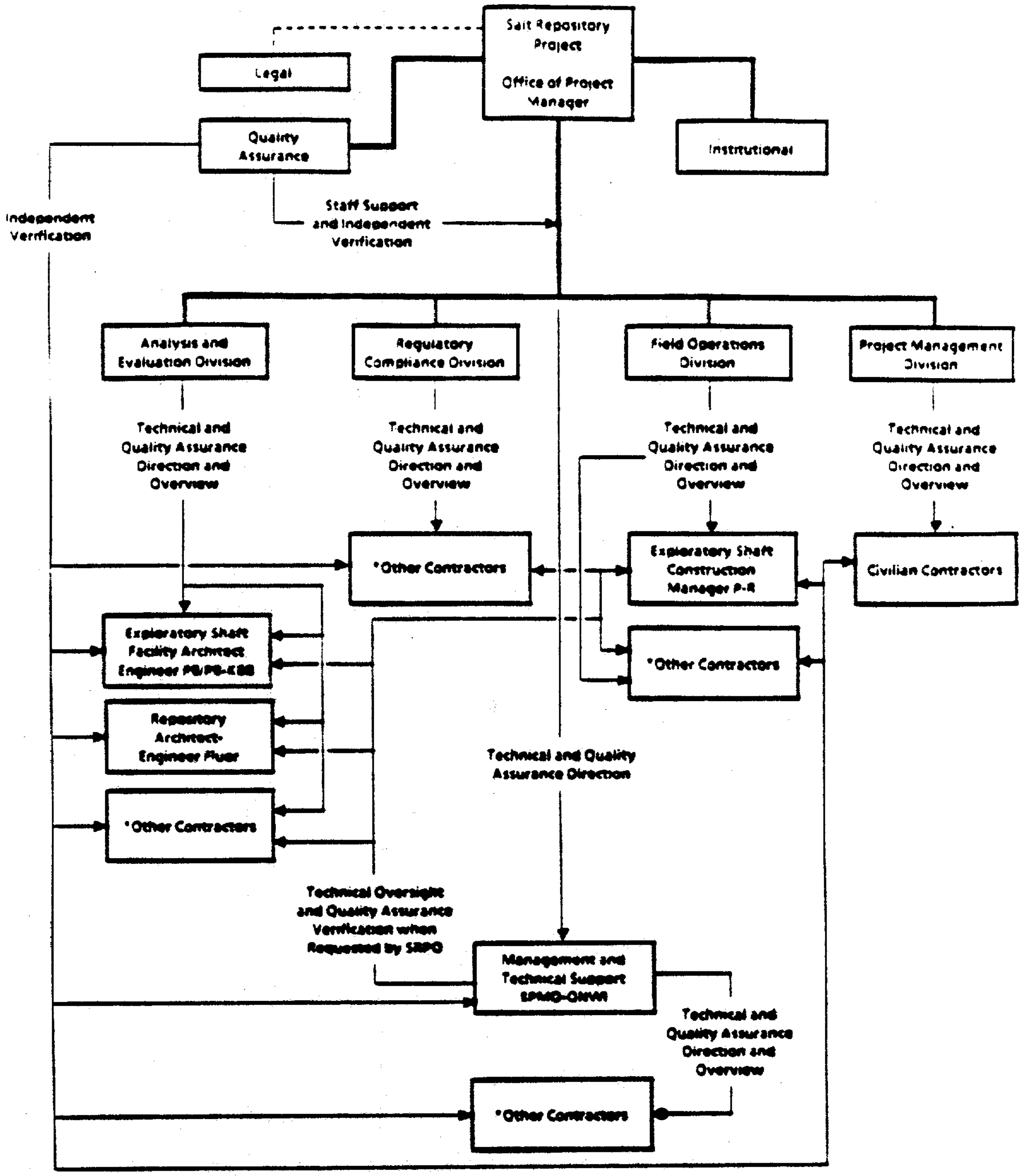
Based on its overall responsibility for defining and ensuring that appropriate quality assurance requirements were established and implemented, and to further ensure uniform and acceptable interpretations of the mandated requirements of the DOE Orders and OGR Quality Assurance Plan and its references, the SRPO prepared the SRPO Quality Assurance Plan (DOE, 1985a). This plan, organized into the 18-requirement structure which coincided with the format and content of ANSI/ASME NQA-1 and 10 CFR Part 50, Appendix B, synthesized and consolidated requirements for the SRPO into a single requirements document. Requirements from the SRPO Quality Assurance Plan and from the OGR Quality Assurance Plan were included in appropriately graded SRPO quality assurance specifications which were contractually imposed on the SRP direct-report contractors, who in turn transmitted appropriate information to their subcontractors. The SRPO Quality Assurance Plan also defined the requirements and responsibilities for the internal SRPO organizations.

Having had the overall management responsibilities for the direction, content, and effective implementation of quality requirements affecting all SRP activities, the SRPO coordinated the quality assurance programs for all the SRP direct-report contractors (see Figure 9-7). Although some work activities and the implementation of certain quality assurance programs applicable to that work had been delegated to contractors, the SRPO retained ultimate responsibility for the delegated functions and activities, and for ensuring that these contractors effectively responded to applicable quality requirements.

Based on the quality requirements stipulated in SRPO-prepared quality assurance specifications, each SRPO contractor established and implemented a quality assurance program except where a contractor was working under the next higher tier organization's quality assurance program. These exceptions included some small, specialized organizations, individual contributors, and educational institutions where the work activity was short-term or where it would not be an effective utilization of project resources to develop and maintain a quality assurance program. Quality assurance programs consisted primarily of programmatic controls over technical work/activities. These controls, which were jointly established by cognizant technical and quality assurance organizations, were implemented by the line organizations that performed the work/activity. Quality assurance program descriptions consisted of a general discussion of the program (plan or manual), administrative quality assurance procedures that provided instructions for implementing and applying applicable quality requirements, and detailed technical procedures that provided instructions for the performance of work (testing, investigations, etc.). Technical procedures incorporated applicable requirements specified in the administrative quality assurance procedures.

The SRPO was responsible for accepting the quality assurance program descriptions and administrative quality assurance procedures of its direct-report contractors while delegating responsibility for preliminary review and disposition to the Integration Support Services Contractor (formerly the Management and Technical Support Contractor). The SRPO was responsible for the review and disposition of the Integration Support Services Contractor quality assurance program and final disposition based on the reviews performed by the ISSC of SRPO's direct-report contractors' documents. Each direct-report contractor was responsible for the acceptance of the quality assurance programs prepared by its subcontractors.

Except for the specific exceptions previously identified, e.g., small, specialized organizations, individual contributors, and educational institutions, not necessarily requiring quality assurance programs, all SRP participants were required to establish and maintain formal internal quality assurance audit programs to monitor implementation of their quality assurance programs.



* Includes Federal Agencies, National Laboratories, Civilian Contractors, Educational Institutions, Consultants

Salt Repository Project Organization

Figure 9-7

Participants who awarded subcontracts to other contractors were required to conduct external audits of the quality assurance programs of the contractors for whom they were responsible. The Integration Support Services Contractor was responsible for supporting the SRPO in scheduling and supporting external audits of other major SRP contractors, integrated contractors, and interagency contractors, while the SRPO audited the ISSC. In addition, SRPO auditors accompanied audit teams of the SRPO direct-report contractors on selected audits to observe audit performance and to evaluate the effectiveness of the contractors' audit processes. The SRPO, in turn, was audited by the DOE/Chicago Operations Office and the OGR at regular intervals.

Deficiencies identified during audits and surveillances were documented and reported to appropriate management. Resulting corrective actions were documented and implemented by the audited organization, and were verified for acceptance and closeout by the auditing organization.

SRP participants (when applicable) were expected to maintain tracking and trending systems that provided sufficient visibility of significant quality problems so that any recurrence would be immediately recognized. In supporting the SRPO for external contractor audits, the ISSC, based on the results of their audits, compiled their results and periodically issued Trend Analysis Reports to SRPO management for their understanding quality assurance programmatic status, and for evaluating and/or investigating indications of adverse quality trends. The most recent report (ONWI-009-87-0067) was issued to the SRPO in January 1988 and included the period July 1986 through June 1987.

A more detailed description of the quality controls in effect during site exploration and those that would have been implemented during site characterization are contained in Section 8.6 of the Consultation Draft Site Characterization Plan (DOE, 1988).

9.2.2 Summary and Status of SRP Contractors Quality Assurance Programs

Listed in this section are summaries of project activities, status of latest quality assurance program documents, audits, and surveillances for each of the SRPO direct-report contractors. Included also are other selected Salt Repository Project contractors/subcontractors whose project activities were designated or otherwise determined to be Quality Level 1 or 2. Quality level determinations were made using the application guidelines from Supplement 8, Quality Assurance Plan for High-Level Radioactive Waste Repositories, OGR/B-3 (DOE, 1986), and the SRPO Quality Assurance Plan (1985).

(1) **Contractor Name/Location: BATTELLE MEMORIAL INSTITUTE, PROJECT MANAGEMENT DIVISION, OFFICE OF NUCLEAR WASTE ISOLATION,** Hereford, Texas

Procurement No: DE-AC02-87CH10290
Highest Quality Level Determined: 1

Title/Summary of Work Activity: INTEGRATION SUPPORT SERVICES. This project required that assistance be provided to the Salt Repository

Project, based on direction from DOE/SRPO, to ensure that project work was adequately defined, planned, budgeted, scheduled, and executed in a structured and systematic manner, and that the work met the project technical schedule, cost, safety, quality, and interface requirements. Also included integrating the work of various project participants such as national laboratories, other Federal agencies, and DOE direct-report contractors.

Contractual Quality Assurance Requirements: SRPO QA requirements included in letter contract dated August 14, 1987.

Status of Quality Assurance Documents: Office of Nuclear Waste Isolation (ONWI) QA Manual dated December 21, 1986, approved by SRPO per letter dated September 5, 1986, requesting incorporation of minor comments. SRPO notified by letter dated January 22, 1987, of resolution of comments. SRPO acceptance of QA procedures pending based on determination by SRPO on extent of necessary reviews.

Summary of Latest Audits (A)/Surveillances (S): SRPO Audit ONWI-87-001-E performed January 1987 resulted in one finding and two observations. SRPO accepted responses to audit June 1987. SRPO Verification Surveillance S-SRPO-87-012-E performed October 1987, one deficiency identified. ONWI response to surveillance transmitted to SRPO December 1987.

Remarks: At project closeout revision to ONWI QA Manual was in preparation to reflect changes resulting from project transition to Texas site office.

(2) **Contractor Name/Location: FLUOR TECHNOLOGY, INC.**, Irvine, California

Procurement No: DE-AC02-83WM46656
Highest Quality Level Determined: 1

Title/Summary of Work Activity: NUCLEAR WASTE REPOSITORY IN SALT. This project required providing the necessary professional design services for the conceptual design, advanced design, and supporting technical activities for a waste repository in salt.

Contractual Quality Assurance Requirements: SRPO QA requirements included in contract dated September 2, 1983; SRPO QA Specification dated September 12, 1986 (not invoked on Fluor); SRPO QA Specification in preparation to include OGR/B-3 requirements.

Status of Quality Assurance Documents: Fluor QA Plan dated February 21, 1984, approved by SRPO June 2, 1986; Fluor QA Manual (procedures) dated November 1, 1985, approved by SRPO June 2, 1986; Fluor transmitted revised procedures October 1987. ONWI transmitted review status and comments on procedures to SRPO December 15, 1987, (ONWI-009-87-035) with conditional acceptance recommendation.

Summary of Latest Audits (A)/Surveillances (S): SRPO Audit FLUOR-87-13-E performed September 1987. ONWI recommended acceptance of responses to identified findings received from Fluor in letter transmitted to SRPO February 1988, along with recommendation that audit be closed.

Remarks: At project closeout revision to SRPO QA Specification was in preparation to include OGR/B-3 (August 1986) requirements.

(3) Contractor Name/Location: **STONE & WEBSTER ENGINEERING CORPORATION (SWEC),** Boston, Massachusetts

Procurement No.: DE-AC02-87CH10285
Highest Quality Level Determined: 1

Title/Summary of Work Activity: **Technical Field Services Contractor.** The primary objectives of this project were the characterization, evaluation, and development of required supporting technologies, for licensing purposes, of a candidate site in salt host rock.

Contractual Quality Assurance Requirements: SRPO QA Specification, dated December 14, 1987, transmitted to SRPO for review and approval on January 13, 1988 (not invoked on SWEC).

Status of Quality Assurance Documents: SWEC QA Plan, dated October 23, 1987, was transmitted to SRPO for review on October 26, 1987. ONWI transmitted review status and comments on the QA Plan to SRPO December 3, 1987 (ONWI-009-87-0355), with conditional acceptance recommendation.

Summary of Latest Audits (A)/Surveillances (S): None performed on this contract.

Remarks: Submittal of TFSC QA Plan by SWEC to be submitted to SRPO on or about March 1, 1988. The following were under subcontract to SWEC after turnover by ISSC/ONWI October 1987, and were transmitted to SRPO February 1, 1988 (SRS.880201004):

The Earth Technology Corporation (SWEC 1700-C-0020). ONWI Audit 86-E-22 performed December 1986 has all corrective actions closed. However, audit remained open pending submittal/acceptance by ONWI of revised topical report "Avery Island Core Aging and Storage Effects Study."

Science Application International Corporation (SWEC 17500-C-0009). QA procedures QP 2.0, QP 9.0, QP 12.0, QP 15.0, and QP 19.0 dispositioned unacceptable by ONWI.

Golder Associates (SWEC 17500-C-0001). SRPO Audit GOLDER-87-002-E performed February 1987. Responses to audit from Golder received and accepted by SRPO.

RE/SPEC Inc. (SWEC 17500-C-0014). ONWI Audits 86-E-16 and 86-E-20 closed by SWEC and addressed in SWEC audit performed November 1987. Audit remained open pending corrective action for CAI-4.

INTERA Technologies, Inc. (SWEC 17500-C-0017). ONWI Audit 87-E-09 performed April 1987. One item was open but completed per INTERA Project Status Report, PSR-15, dated November 1987.

Research & Planning Consultants (SWEC 17500-C-0010). ONWI Surveillance 87-S-14 performed August 1987. Corrective actions for two deficiencies identified complete. Audit remained open pending verifications; however, SWEC recommended closing based on review/acceptance of submitted records.

Harmsworth Associates (SWEC 17500-C-0007). QA Procedure TP-01 dispositioned unacceptable by ONWI; however, SWEC recommended closing based on review/acceptance of submitted records.

Engineering Science, Incorporated (SWEC 17500-C-0007). Project Management Procedures PMP-9 and PMP-10 dispositioned unacceptable by ONWI.

University of California-Riverside (SWEC 17500-C-0019). ONWI Surveillance 87-S-16 performed March 1987. Response to identified deficiency conditionally accepted October 1987 pending verification. SWEC recommended closing without verification.

SWEC reviewed the QA Plans from all assigned contractors and had no major problems; however, final QA program acceptance not made by SWEC.

(4) **Contractor Name/Location:** **U.S. BUREAU OF MINES (USBM)**, Pittsburgh, Pennsylvania

Procurement No.: DE-A197-86WM46663
Highest Quality Level Determined: 1

Title/Summary of Work Activity: **TECHNICAL SUPPORT.** Objective of the project was to provide SRPO project management/coordination in mining/drilling techniques, technical review and guidance, and technical studies related to nuclear waste isolation.

Contractual Quality Assurance Requirements: SRPO QA requirements included in QA Specification, dated November 6, 1986.

Status of Quality Assurance Documents: ONWI transmitted review status on the Project QA Plan, Revision 0, to SRPO December 30, 1987, with acceptance recommendation.

Summary of Latest Audits (A)/Surveillances (S): None performed on this procurement.

Remarks: SRPO QA Specification for USBM did not include OGR/B-3 (August 1986) requirements.

(5) Contractor Name/Location: **OAK RIDGE OPERATIONS/SCIENCE APPLICATIONS INTERNATIONAL CORPORATION**, Oak Ridge, Tennessee/Columbus, Ohio

Procurement No.: DE-AC05-81OR20837
Highest Quality Level Determined: 2

Title/Summary of Work Activity: **SRP Regulatory Technical Information Support.** The project involved providing assistance in the development of the Site Characterization Plan, identifying waste management licensing issues, developing issues resolution strategies, establishing information needs, and supporting the implementation of the SRP issues tracking system.

Contractual Quality Assurance Requirements: SRPO QA requirements included in QA Specification dated September 11, 1986.

Status of Quality Assurance Documents: QA Plan/Procedures not required as work was performed under SRPO QA program including SRPO procedures. However, SAIC elected to submit QA Plan, Revision 0 (July 31, 1987) to SRPO for review and acceptance. Transmitted to ONWI for review October 13, 1987 (252-87-RC). Review not completed because of project termination.

Summary of Latest Audits (A)/Surveillances (S): SRPO Audit SAIC-86-2-E performed May 5-6, 1986, in Columbus, Ohio. Audit closed.

Remarks: SAIC QA Plan, Revision 0 submitted to SRPO not required.

(6) Contractor Name/Location: **OAK RIDGE NATIONAL LABORATORY**, Oak Ridge, Tennessee

Procurement No.: DE-AC05-84OR21400
Highest Quality Level Determined: 1

Title/Summary of Work Activity: **SENSITIVITY AND UNCERTAINTY ANALYSIS METHODS DEVELOPMENT.** The objective of this project is to develop a computer-automated sensitivity analysis methodology to be used to produce sensitivity versions of selected performance assessment computer codes.

Contractual Quality Assurance Requirements: SRPO QA requirements included in QA Specification dated September 11, 1986.

Status of Quality Assurance Documents: QA Plan dated December 1986 and implementing procedures were approved by SRPO July 31, 1987.

Summary of Latest Audits (A)/Surveillances (S): SRPO Audit ORNL-87-16-E performed October 1987 identified two deficiencies. ONWI recommended acceptance of responses received from ORNL in a transmittal letter (SRW.871228.0016) to SRPO December 1987.

Remarks: None.

(7) Contractor Name/Location: **ARGONNE NATIONAL LABORATORY**, Argonne, Illinois

Procurement No.: 109-ENG38-109
Highest Quality Level Determined: 1

Title/Summary of Work Activity: **INTERACTION OF WASTE PACKAGE MATERIALS WITH SIMULATED SALT REPOSITORY ENVIRONMENTS.** The objective of this project was to provide the identification and qualification of the important variables and processes involved in the corrosion of metals being proposed for the containment of high-level nuclear waste in salt rock environments.

Contractual Quality Assurance Requirements: SRPO QA requirements included in QA Specification dated September 15, 1986.

Status of Quality Assurance Documents: QA Plan dated May 1987 approved by SRPO November 15, 1987. All QA procedures approved by SRPO November 3, 1987.

Summary of Latest Audits (A)/Surveillances (S): No Audits/Surveillances performed due to late development/implementation or required QA program.

Remarks: None.

(8) **Contractor Name/Location**: **ARGONNE NATIONAL LABORATORY**, Argonne, Illinois

Procurement No.: 109-ENG38-109
Highest Quality Level Determined: 1

Title/Summary of Work Activity: Two (2) projects are identified: **(a) TECHNICAL REVIEW AND (b) MULTIDISCIPLINARY PEER REVIEW OF DOE-CONTRACTOR DOCUMENTS.** These projects supported SRPO by the performance of (a) technical reviews to satisfy minor milestones established within the SRP, and (b) multidisciplinary critiques by ANL and selected external personnel of selected SRPO. Contractor documents to satisfy major milestones within the SRP.

Contractual Quality Assurance Requirements: SRPO QA requirements included in SRPO QA Specifications dated (a) September 18, 1986, and (b) September 17, 1986.

Status of Quality Assurance Documents: (a) QA Plan dated September 8, 1986, was conditionally accepted by SRPO February 12, 1987, and (b) QA Plan dated August 12, 1986, accepted August 27, 1986. QA procedures for (a) not required as ANL was using SRPO QA procedures, and (b) QA procedures included in QA Plan as Appendices.

Summary of Latest Audits (A)/Surveillances (S): SRPO Audit ANL-87-10-E performed May 1987 included both projects. Three deficiencies identified for (a) and one for (b). Responses to audit received and closed December 9, 1987. SRPO Surveillance S-ANL-87-09-E performed October 2, 1987, with no deficiencies identified.

Remarks: None.

(9) **Contractor Name/Location:** **LAWRENCE BERKELEY LABORATORY (LBL),** Berkeley, California

Procurement No.: DE-AC03-76SF00098
Highest Quality Level Determined: 1

Title/Summary of Work Activity: **SOLUBILITIES AND SPECIATION OF RADIONUCLIDES IN BRINE.** This project required providing reliable and creditable data and documentation for assessing the desirability of a license for the long-term storage of radionuclides in a salt repository.

Contractual Quality Assurance Requirements: SRPO QA requirements included in SRPO QA Specification dated September 8, 1986.

Status of Quality Assurance Documents: LBL Project Quality Assurance Plan, dated January 28, 1987, approved by SRPO April 17, 1987.

Summary of Latest Audits (A)/Surveillances (S): SRPO Audit LBL-87-006-E conducted December 14, 1987. Six deficiencies identified; SRPO accepted five responses but required project status reports to be submitted to close out the sixth deficiency (SRW.871119.0021). Project status reports were submitted through August 1987. Audit considered to be closed.

Remarks: Revision to SRPO QA specification for LBL in preparation at project closeout to include OGR/B-3 requirements.

(10) **Contractor Name/Location:** **PARSONS BRINCKERHOFF/PB-KBB,** Houston, Texas

Procurement No.: DE-AC97-86WM46664
Highest Quality Level Determined: 1

Title/Summary of Work Activity: ACTIVITIES IN SUPPORT OF DESIGN OF THE EXPLORATORY SHAFT FACILITY. This project required providing professional services for project management, integration engineering, and to provide Title II and III design efforts.

Contractual Quality Assurance Requirements: SRPO QA requirements included in SRPO QA Specification dated September 12, 1986 (Revision 01).

Status of Quality Assurance Documents: Quality Assurance Plan (ES-346-01), Rev. 01, approved by SRPO August 20, 1987. Procedures (QA and Technical) dated November 1987 (Revision 4 ES-347-01) approved by SRPO August 28, 1987.

Summary of Latest Audits (A)/Surveillances (S): SRPO Surveillance S-PB/PB-KBB-88-01 performed December 8-9, 1987, to followup and verify corrective actions proposed by PB on previous audits and the SRPO-issued Corrective Action Report. As a result SRPO Audit No. PB-86-3-E, SRPO Audit No. PB/PB-KBB-87-07-E, SRPO Surveillance S-PB/PB-KBB-87-03-E, Corrective Action Report 01-PB, and S-PB/PB-KBB-88-01 were closed.

Remarks: Revision to SRPO QA Specification for PB/PB-KBB in preparation at project closeout to include OGR/B-3 requirements.

(11) **Contractor Name/Location:** **PARSONS REDPATH,** Hereford, Texas

Procurement No.: DE-AC02-83CH10125
Highest Quality Level Determined: 1

Title/Summary of Work Activity: CONSTRUCTION MANAGER - EXPLORATORY SHAFT FACILITY IN SALT. This project required construction manager support for an exploratory shaft facility in salt, and to furnish all things necessary or convenient to accomplish the construction. No design responsibility was required.

Contractual Quality Assurance Requirements: SRPO QA requirements included in SRPO Contract of November 1982 for NQA-1-79. Contract subsequently revised to include NRC Review Plan.

Status of Quality Assurance Documents: Quality Assurance Plan, dated October 1986, approved by SRPO November 21, 1986. QA Procedures, dated January 9, 1987, and Quality Assurance Manual, January 21, 1987, conditionally approved.

Summary of Latest Audits (A)/Surveillances (S): SRPO Audit No. PR-88-01-E performed November 2-3, 1987, resulted in nine deficiencies, but audit report not issued due to project closing.

Remarks: None.

(12) Contractor Name/Location: PACIFIC NORTHWEST LABORATORY, Richland, Washington

Procurement No: DE-AC06-76-RL01830
Highest Quality Level Determined: 1

Title/Summary of Work Activity

- **WASTE PACKAGE PROGRAM (WPP).** The objective of the project was to develop the performance models and supporting data base on package components and system interactions to meet performance objectives. Major tasks include waste form evaluation and testing; West Valley Waste Glass; environmental studies; metal barrier testing; integrated testing; and waste package component modeling defense waste glass studies; and SCP support.

- **DISEQUILIBRIUM STUDY OF NATURAL RADIONUCLIDES OF U AND Th SERIES IN BRINEY GROUND WATERS AND CORES 1 (DS).** The objective of this project was to employ the technology of disequilibrium of the natural U-238 and Th-232 decay series to provide information in determining selected radionuclide transport parameters, and in the identification of aquifer source regions at a potential nuclear waste repository site. Studies include natural radionuclides of U/Th decay chain; REE, other cations and anions, and fluid inclusions; laboratory investigations of field observations.

- **MATERIALS CHARACTERIZATION CENTER (MCC).** The objective of this project was to develop experimental techniques and instrumentation for the measurement of pH and Eh in saturated brines under conditions of high temperature and pressure; development of SRP test methods.

- **PERFORMANCE ASSESSMENT (PA).** The objective of this project was to provide performance assessments of activities for total system performance assessment and site performance assessment. Specific tasks include code development/improvement, verification, benchmarking and documentation support; system and simulation analysis; salt site hydrologic assessment support; and geologic events and process analysis.

Contractual Quality Assurance Requirements:

WPP - SRPO QA requirements included in SRPO QA Specification dated September 8, 1986. SRPO QA Specification dated December 15, 1987, including OGR/B-3 requirements forwarded SRPO on January 12, 1988.

DS - SRPO QA requirements included in SRPO QA Specification dated September 8, 1986. Revision to SRPO QA Specification including OGR/B-3 requirements, in preparation.

MCC - SRPO QA requirements included in SRPO QA Specification dated September 11, 1986. SRPO QA dated December 16, 1987, including OGR/B-3 requirements forwarded SRPO January 25, 1988.

PA - SRPO QA requirements included in SRPO QA Specification dated September 11, 1986. SRPO QA Specification dated December 16, 1987, including OGR/B-3 requirements forwarded SRPO January 25, 1988.

Status of Quality Assurance Documents:

WPP - QA Plan (WCT-003 Revision 1) approved by SRPO November 3, 1987.

DS - QA Plan (WCT-008 Revision 1) approved by SRPO November 3, 1987.

MCC - QA Plan (WCT-002 Revision 1) approved by SRPO November 3, 1987.

PA - QA Plan (WCT-004 Revision 1) approved by SRPO September 4, 1987.

Summary of Latest Audits (A)/Surveillances (S):

Audit PNL-87-021-E performed on October 6-8, 1987. Audit covered Performance Assessment, and Disequilibrium Studies Projects. Four findings and three observations were recorded. PNL's proposed corrective actions sent to SRPO February 22, 1988, and ONWI February 24, 1988.

Surveillance S-PNL-87-004-E performed on March 17-18, 1987. Resolution of deficiencies noted, except bullet #1 of the report, have been accepted. SRPO response transmitted to DOE Richland on December 2, 1987.

Remarks: None.

(13) **Contractor Name/Location: LAWRENCE LIVERMORE NATIONAL LABORATORY (LLNL)**, Livermore, California

Procurement No.: W-7405-ENG-48
Highest Quality Level Determined: 1

Title/Summary of Work Activity: EQ3/EQ6 GEOCHEMICAL MODELING. The objective of this project was to develop a geochemical modeling code to make estimates of radionuclide solubilities.

Contractual Quality Assurance Requirements: SRPO QA requirements included in SRPO QA Specification, dated September 11, 1986; SRPO QA Specification in preparation to include OGR/B-3 requirements.

Status of Quality Assurance Documents: LLNL Quality Assurance Program Plan (QAPP) consists of requirements and procedures of various dates and revisions; SRPO transmitted review status of LLNL's QAPP to LLNL on May 5, 1987 (140-87-AE), and May 19, 1987 (102-87-AE), with conditional acceptance recommendation. LLNL transmitted their draft project specific (SRPO) QA Plan to SRPO December 22, 1987. Review not performed.

Summary of Latest Audits (A)/Surveillances (S): SRPO Audit LLNL-87-04-E performed April 27 through May 1, 1987. LLNL transmitted responses to findings 2 and 3 to SRPO December 16, 1987 (I-1228-87-014).

Remarks: SRPO transmitted Corrective Action Report CAR-03-LLNL September 28, 1987.

(14) **Contractor Name/Location:** **BROOKHAVEN NATIONAL LABORATORY,** Upton, New York

Procurement No.: DE-AC02-76CH0016
Highest Quality Level Determined: 1

Title/Summary of Work Activity: **RADIATION EFFECTS STUDIES ON ROCK SALT.** The objective of this project was to provide data and understanding of the effects of radiation, emanating from a waste package, on surrounding rock salt.

Contractual Quality Assurance Requirements: SRPO QA requirements included in SRPO QA Specification dated September 8, 1986.

Status of Quality Assurance Documents: BNL QA Plan dated October 27, 1987, approved by SRPO on October 27, 1987. QA Procedures approved by SRPO on various dates, latest is BRE 1.6C on January 15, 1988.

Summary of Latest Audits (A)/Surveillances (S): SRPO Audit BNL-88-02-E performed December 1987. No deficiencies identified. Audit closed. Corrective actions from previous SRPO Audit BNL-87-05-E verified and closed as a result of latest audit.

Remarks: None.

(15) **Contractor Name/Location:** **U.S. GEOLOGICAL SURVEY,** Reston, Virginia/ Denver, Colorado

Procurement No.: DE-AC197-84WM46658
Highest Quality Level Determined: 1

Title/Summary of Work Activity: **SALT RADIATION TECHNICAL STUDIES and TECHNICAL REVIEW SUPPORT.** The objective of this project was to study radiation effects on salt; however, laboratory analysis activities were never fully implemented due to budgetary and QA Program problems. Independent technical reviews were performed under SRPO QA Program requirements.

<u>Contractual Quality Assurance Requirements</u>: SRPO QA requirements included in SRPO QA Specification dated December 16, 1986. Revised SRPO QA Specification for USGS was in preparation at project closeout to include OGR/B-3 requirements.

<u>Status of Quality Assurance Documents</u>: USGS QA Program Plan USGS-QAPP-01, Revision 0 rejected by SRPO June 1987. Eighteen QA procedures conditionally accepted June 1987. No revisions to QA program documents performed.

<u>Summary of Latest Audits (A)/Surveillances (S)</u>: No audits performed. SRPO Surveillance S-USGS-87-007-E performed June 1987. One deficiency identified. No reply from USGS at project closeout.

<u>Remarks</u>: SRPO Surveillance S-USGS-87-007-E remained open at project closeout.

(16) **<u>Contractor Name/Location</u>: UNC GEOTECH (formerly UNC Technical Studies Services, Inc.)**, Grand Junction, Colorado

<u>Procurement No.</u>: DE-AC07-76GJ0166A
<u>Highest Quality Level Determined</u>: 1

<u>Title/Summary of Work Activity</u>: GEOCHEMICAL ASSISTANCE PROJECT. The objective of this project was the performance of various chemical, geochemical, and other laboratory studies on the properties of minerals and rock salt. Additionally, work included multispectral remote sensing, and aeromagnetic mapping and interpretive reporting.

<u>Contractual Quality Assurance Requirements</u>: SRPO QA requirements included in SRPO QA Specification dated September 11, 1986; revision to SRPO QA Specification to include OGR/B-3 requirements was in preparation at project closeout.

<u>Status of Quality Assurance Documents</u>: UNC QA Manual dated March 31, 1987, and QA Program Plan 84-2, Revision 7, including QA procedures accepted by SRPO January 29, 1987. A revision to the QA Manual dated November 15, 1987, transmitted to SRPO January 7, 1988, for review. Review was not performed because of project termination.

<u>Summary of Latest Audits (A)/Surveillances (S)</u>: SRPO Surveillance S-UNC-87-002-E performed January 1987. One deficiency identified, response to deficiency accepted by SRPO; however, surveillance remained open pending a revised procedure. SRPO Audit UNC-87-7-E performed April 1987. Four deficiencies identified. Responses for two deficiencies accepted by SRPO, two others submitted to SRPO December 23, 1987; review was not performed because of project termination. Audit remains open.

Remarks: Both Surveillance and Audit remained open at project closeout.

(17) Contractor Name/Location: UNIVERSITY OF TEXAS AT AUSTIN, BUREAU OF ECONOMIC GEOLOGY, Austin, Texas

Procurement No.: DE-AC97-83WM46651
Highest Quality Level Determined: 1

Title/Summary of Work Activity: GEOLOGIC STUDIES OF BEDDED SALT DEPOSITS. The objective of this project was to study bedded salt properties; perform field studies; and perform as core curator for the SRP which included geochemical laboratory analysis, subsurface field investigations, and hydrology and modeling studies.

Contractual Quality Assurance Requirements: SRPO QA requirements included in SRPO QA Specification dated October 23, 1986; revision to SRPO QA Specification in preparation to include OGR/B-3 requirements.

Status of Quality Assurance Documents: TBEG QA Program dated January 25, 1985, and QA Plan dated November 21, 1985, approved by SRPO. Both documents revised and combined, then transmitted as Revision 6 to SRPO on December 21, 1987. QA procedures are in various stages of revision/review or have been accepted by SRPO; ten procedures have been accepted, seven are being revised, and one was transmitted to SRPO December 21, 1987. Review of the procedure has not been performed as of this date.

Summary of Latest Audits (A)/Surveillances (S): SRPO Audit TBEG-86-8-E performed June 1986 is still open. Five deficiencies identified. Proposed corrective actions received by SRPO; two accepted, one requires additional information, two in process because of procedure revisions. SRPO Audit TBEG-87-15-E performed July 1987. Nine deficiencies identified. Seven corrective action responses accepted by SRPO, two require additional information and resubmittal to SRPO. Concurrence with TBEG proposed schedule for completion of open items for TBEG-86-8-E transmitted to SRPO December 16, 1987. Summary of corrective action responses for TBEG-87-15-E transmitted to SRPO on December 18, 1987. SRPO Surveillance S-TBEG-87-1-E performed November 1986; two deficiencies identified and corrective action responses accepted by SRPO. Surveillance remains open pending SRPO receipt of revised procedures.

9.3 RECORDS AND DATA SYSTEMS

The DOE required that the SRP maintain a comprehensive technical information support system to meet the documentation requirements for licensing; for interactions with the host State, the Nuclear Regulatory Commission (NRC), and the public; and as an ongoing tool to support technical efforts throughout the project. The Salt Repository Project Office of DOE delegated the development and maintenance of the system to the Management and Technical Support Contractor (Battelle Project Management Division (BPMD) -

Office of Nuclear Waste Isolation (ONWI)). This system addressed project needs for technical data (raw/unanalyzed data, analyzed/verified data, baselined data), project and quality assurance records, technical literature, and other support documentation. These data and records were required to be collected, organized, extracted, indexed, and stored, and be readily retrievable by a diverse group of users. The procedures and methods of data handling were designed to meet the strictest requirements of quality assurance and configuration management, and all data in the system must be technically defensible.

9.3.1 Background

Requirements for the technical information support system and guidance in the design and implementation have been provided by the SRPO since 1978. During the past 9 years, the overall system and its subsystems have evolved to meet the changing SRP needs for information management. In addition, computer technology has advanced very rapidly during this period, providing improved and cost-effective ways to store and retrieve data and information. BPMD/ONWI took advantage of the new technology and maintained state-of-the-art information systems.

The systems continued to evolve due to dynamic SRP requirements inherent in a multi-phased project such as the SRP. As the project moved toward site characterization, it was anticipated that regulations affecting information management would become even more stringent. The SRPO guidance emphasized the need for an integrated information system that can provide traceability of data from published reports to the field/laboratory source data. Records management and issues tracking are other integral parts of this system. The result was the emerging SRPO Integrated Data Management System (IDMS).

9.3.2 IDMS Overview

The following presents an overview of the SRP Integrated Data Management System (IDMS) and its components which when fully in place would provide access to all project data in whatever format it was collected. The primary objective of the system was to store and preserve for easy retrieval all information necessary for characterization and licensing of the site.

The integrated system as it is being implemented up to this time is composed of four major modules of the IDMS, each of which contains several components:

- Automated Records System (ARS) - A merger of the Records and Information System (RIS), the Mail Log, the Records Turnover Package (RTP), the Reference Tracking System (RTS), and the Reports Distribution Center (RDC) Inventory

- Technical Data Management System (TDMS) - A linking of the SRP Technical Data Base (SRP/TDB), the Sample Inventory Management System (SIMS), the Detail Data Reference System (DDRS), and the Field Data Analysis System (FDAS)

- Contract Data Management System (CDMS) - A merger of the Deliverables Tracking System (DTS), the Catalogue, and the Report Clearance System (RCS)

- Administrative Tracking System (ATS) - A linking of the Commitment/Action Tracking System (C/ATS), the Issues Tracking System (ITS), the Document Comments Tracking and Response System (DCTRS), and the Reports Mailing List.

All working modules could be accessed through menus that guide users to the kind of information or data they are seeking. All units of the IDMS reside on a common computer and utilize the same software.

For information relating to the component programs that comprise the IDMS, refer to Report BMI/ONWI-635, Salt Repository Project Information Systems Description (ONWI, 1987).

9.3.3 SRP Records Turnover

Project records include the documents generated during the life of a given contract or subcontract, including laboratory/field notebooks, test results, communications, and reports, except for the official procurement documents. The project records packages vary in size from very small (5 to 10 documents) to very large (10,000 documents) and include those generated by DOE prime contractors, national laboratories, Federal agencies, educational institutions, consultants, and supporting subcontractors participating in the SRP. Records are turned over at the completion of a contract or periodically during the contract, as specified for long-term contracts.

Since 1979, approximately 300 packages from 144 contracts have been turned over to BPMD/ONWI by contractors and subcontractors and processed for storage and retrieval. These efforts have resulted in a data bank of more than 500,000 records which can be searched and retrieved on line via the ARS.

The RTP (Records Turnover Package) data are organized by packages associated with a contract or work package number and can be searched by 21 data elements. Each package has one "control" record which contains a unique identification number, the contract name, the title, the contract start and end dates, and the project manager information. The remainder of the package is comprised of records corresponding to project-related documents such as purchase orders, technical papers and reports, memoranda, well logs, computer printouts, negatives, etc. Each record contains the document title, document date, microfilm number, and the unique identification number for that package. The document code, as in the ARS, provides a precise description of the document (incoming letter, project status report, document review form, etc.). The microfilm number is the retrieval mechanism, representing a roll and frame number where the document can be viewed on microfilm and also a file-drawer location where a hard copy of the document can be found. Each record of a package is linked to the "control" record by the unique identification number.

9.4 CHAPTER 9 REFERENCES

ASME (American Society of Mechanical Engineers), 1986. Quality Assurance Program Requirements for Nuclear Facilities, ANSI/ASME NQA-1-1986, New York, NY.

BPMD (Battelle Project Management Division), 1988. Trend Analysis Report - BPMD Quality Assurance Program, ONWI-009-87-0067, Battelle Memorial Institute, Hereford, TX, January 29.

DOE (U.S. Department of Energy), 1984, revised 1986. Office of Geologic Repositories Quality Assurance Plan for High-Level Radioactive Waste Repositories, DOE/RW-0095, Office of Civilian Radioactive Waste Management, Washington, DC.

DOE (U.S. Department of Energy), 1985a. Quality Assurance Plan, Revision 0, Salt Repository Project Office, Columbus, OH.

DOE (U.S. Department of Energy), 1985b. Quality Assurance Management Policies and Requirements, DOE/RW-0032, Office of Civilian Radioactive Waste Management, Washington, DC.

DOE (U.S. Department of Energy), 1988. Site Characterization Plan, Deaf Smith County Site, Texas, Consultation Draft, Nuclear Waste Policy Act (Section 113), DOE/RW-0162, U.S. Department of Energy, Washington, DC.

NRC (U.S. Nuclear Regulatory Commission), 1984. NRC Review Plan: Quality Assurance Programs for Site Characterization of High Level Nuclear Waste Repositories, compiled by Division of Waste Management, Office of Nuclear Material Safety and Safeguards, Washington, DC.

ONWI (Office of Nuclear Waste Isolation), 1982. Office of Nuclear Waste Isolation FY 82 Technical Project Plan, ONWI-19(FY 84), Battelle Memorial Institute, Columbus, OH.

ONWI (Office of Nuclear Waste Isolation), 1987. Salt Repository Project Information Systems Description, BMI/ONWI-635, Battelle Memorial Institute, Columbus, OH.

SRPO (Salt Repository Project Office), 1987. Salt Repository Project Networks, ONWI Controlled Document, Office of Nuclear Waste Isolation, Battelle Memorial Institute, Columbus, OH.

FEDERAL REGULATIONS (By Number)

10 CFR Part 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants, 1981.

10 CFR Part 60, Disposal of High-Level Radioactive Wastes in Geologic Repositories, 1986.

DOE ORDERS

DOE 2250.1, Cost/Schedule Control Systems Criteria, 1983.

DOE 5700.6A, Quality Assurance, 1981. (Incorporated into and replaced by DOE 5700.6B, 1986.)

DOE 5700.6B, Quality Assurance, September 23, 1986.

DOE 5700.7, Field Work Package Proposal and Authorization System, 1981.

DOE 5700.7B, Work Authorization Systems, 1987.