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THE TECHNOLOGY OF HIGH-LEVEL NUCLEAR WASTE DISPOSAL

Advances in the Science and Engineering of the Management of High-Level Nuclear Wastes

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Peter L. Hofmann, Editor

Battelle Memorial Institute
Project Management Division

Volume 2, 1982

Prepared for the U. S. Department of Energy
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Foreword

This is the second volume of papers in a series of annual volumes dealing with the management of high-level radioactive wastes. The work treated in this collection of papers was performed as part of the Geologic Repository Deployment (GRD) Program (formerly known as the National Waste Terminal Storage Program) of the U. S. Department of Energy.

The objectives of this program are to develop the technology and to provide the facilities for the permanent isolation of high-level and transuranic nuclear wastes resulting from the commercial production of electric power. Major emphasis is on disposal of these wastes in deep geologic formations. This program is unique in its concern for the health and safety of present and future generations.

The Department of Energy is making extensive efforts to inform interested parties about the conduct of its national program. Such efforts include consultation with state governors and state, tribal, and local officials to secure their participation in the resolution of repository siting and other issues. Local and national public information meetings are held, and significant programmatic documents are made readily available. Independent reviews of the GRD Program with respect to both its technical quality and its responsiveness to public concerns are solicited.

We present this second volume of the Advances in the Science and Engineering of the Management of High-Level Nuclear Wastes series with the continued hope that it will contribute to a better understanding of the GRD Program.

J. William Bennett
Acting Associate Director
Office of Geologic Repository Deployment
Office of Civilian Radioactive Waste Management
U.S. Department of Energy

Preface

This is the second of a series of annual volumes dealing with advances in the technology for the disposal of high-level and transuranic nuclear wastes. Volume 1 of this series was released for publication in the summer of 1983.

Papers for Volume 2 were selected from those presented at the National Waste Terminal Storage Information Meeting held in Columbus, Ohio, in the winter of 1980. The papers were subsequently expanded and brought up to date to represent the status of technology that existed in 1981 and 1982.

The twenty papers in this volume are divided into three parts:

Part I—Site Exploration and Characterization
Part II—Repository Development and Design
Part III—Waste Package Development and Design

It is the aim of this series to provide a comprehensive coverage of high-level nuclear waste disposal topics and to serve as a historical record. For this reason, two articles on socioeconomic issues, which reflect the key current interest in institutional questions, are included in Volume 2. Furthermore, although most of the current investigations center on mined geologic disposal of nuclear wastes, work is also proceeding on subseabed disposal. Accordingly, one chapter of this volume is devoted to a review of this technology.

As was the case for Volume 1, every contribution to Volume 2 was reviewed by members of the Editorial Review Board formed for the series. Detailed editing was provided by Maggie Jared and James L. Littlepage of the Department of Energy, Office of Scientific and Technical Information, Technical Information Center.

Finally, I would like to acknowledge the help provided by Jean McLean with the many details involved in this undertaking.

P. L. Hofmann, Editor

The Technology of High-Level Nuclear Waste Disposal
Volume 1, 1981

Peter L. Hofmann, Editor, and **John J. Breslin,** Associate Editor
Battelle Memorial Institute, Project Management Division

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Contents

Part I Site Exploration and Characterization

A Method for Screening the Nevada Test Site and Contiguous Areas for Nuclear Waste Repository Locations

Cont. 10/

S. Sinnock, J. A. Fernandez, J. T. Neal, H. P. Stephens,* and B. L. Hartway†; *Sandia National Laboratories, Albuquerque, NM; †Los Alamos Technical Associates, Inc., Los Alamos, NM

This paper outlines the general concepts of a technical method for systematic screening of the Nevada Test Site (NTS), Nye County, Nevada, for potentially suitable nuclear waste repository locations. After a general discussion of the organization and the purpose of the current screening activity, the paper addresses the steps of the screening method. These steps include:

- **Hierarchically organizing technical objectives for repository performance (an objectives tree)**
- **Identifying and mapping pertinent physical characteristics of a site and its setting (physical attributes)**
- **Relating the physical conditions to the objectives (favorability curves)**
- **Identifying alternative locations and numerically evaluating their relative merits**
- **Investigating the effects of subjective judgments on the evaluations (sensitivity analyses)**
- **Documenting the assumptions, logic, and results of the method**

19 ref, 10 fig.

Organization

The Department of Energy's (DOE) Nevada Nuclear Waste Storage Investigations (NNWSI) Project Office is formally responsible for evaluating the suitability of the NTS for a mined repository that would be constructed deep underground to isolate commercial spent nuclear fuel or high-level radioactive waste (U. S. Department of Energy, 1980b, 1980c, and 1982). The NNWSI are managed by the Nevada Operations Office (NV) of the DOE. Technical support is provided by Sandia National Laboratories (SNL), Los Alamos National Laboratory (LANL), Lawrence Livermore National Laboratory (LLNL), the U. S. Geological Survey (USGS), and Westinghouse Corporation. The NNWSI are part of the DOE's National Waste Terminal Storage Program (NWTS) which is charged with managing and eventually disposing of wastes from the nation's commercial nuclear activities.

Organization and use of specific screening information will be coordinated by the NNWSI Technical Overview Contractor (Fig. 1). Continual review and guidance will be provided by the NNWSI Site Evaluation Working Group (SEWG). The SEWG will evaluate screening results and recommend future characterization options to the NNWSI Site Evaluation Steering Committee. The Steering Committee, in turn, will recommend a specific course of action regarding repository siting activities at the NTS to the Manager of the Nevada Operations Office.

Purpose

The purpose of the NNWSI screening activity is to formally identify geographic locations at the NTS that,

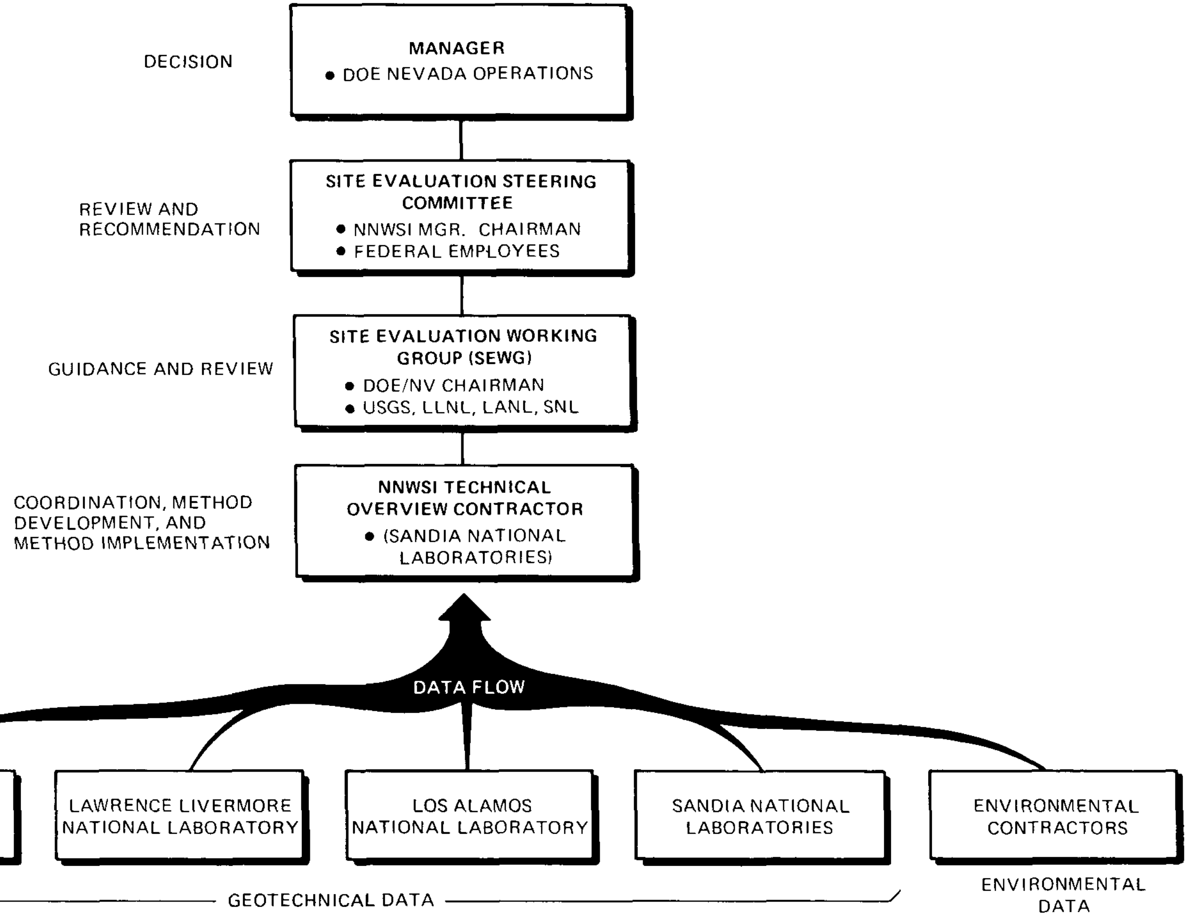

Figure 1 Organization of the NNWSI location screening activity.

based on current information, merit further characterization and evaluation as candidates for a commercial nuclear waste repository. Siting activities currently are limited to a region encompassing the southwest portion of the NTS (Fig. 2) so that interference with the NTS prime mission, nuclear weapons testing (U. S. Department of Energy, 1980b) is avoided.

The proposed screening method will not assess potential repository locations for absolute suitability. Rather, it will identify where potentially suitable locations exist, and it will simultaneously compare the relative merits of the various locations. To presume the capability for assessing suitability on an absolute basis is, in the view of the NNWSI Technical Overview Contractor, both premature and unwise. Two key factors contributing to this view are the lack of absolute standards based on health consequences for each of the many repository siting factors currently documented by various organizations (U. S. Department of Energy, 1980b, 1981a, and 1982); U. S. Department of Energy et al., 1980; International Atomic Energy Agency, 1977; National Academy of Sciences, 1978; Johnson and Deju, 1979; Cochran and Dimitri, 1979; Brunton and McClain, 1977; U. S. Nuclear Regulatory Commission, 1980 and 1981; and the fact that comprehensive site data and engineering designs required for reliable consequence analysis are not yet available. Only after detailed site characterization following location selection will information be adequate to support safety assessments in terms of absolute and regulatory requirements. The current phase of screening will provide the rationale for focusing exploration and characterization efforts on the location or locations that, based on the best judgments of those involved, ultimately will prove suitable to safeguard public health and safety and environmental integrity.

Some decision risk is unavoidable. Nonetheless the screening method outlined here, when properly supported by a broad range of technical disciplines, will reduce the chances that subsequent characterization efforts will be expended on an unsuitable location.

Screening will be accomplished by dividing the siting problem into three components—performance objectives, physical attributes, and favorability functions (criteria); evaluating each component separately; and recombining the components to achieve a comprehensive comparison of alternative locations. This screening process will offer the fundamental advantages that, as each factor is individually considered for any given location, investigators will reduce the possibility that an unacceptable flaw will be overlooked; and, conversely, as the siting factors are recombined to rate alternative locations, investigators will reduce the chances that a mitigating circumstance for an apparently unacceptable flaw will be overlooked and a suitable location needlessly rejected. [For a detailed discussion of the decision-making methodology of the screening method see Warfield (1974), Starr and Zelen (1977), and Kenney and Raiffa (1976).]

The screening activity will rely on data from past and current NNWSI exploratory field work and laboratory studies as well as on other readily available information. Results will be documented in a Screening Summary and Recommendation Report that will provide a basis for deciding whether to proceed with further repository siting studies on the NTS. The report also will show where exploration should be concentrated—both geographically and, perhaps, topically—if the decision is made to proceed with further siting studies.

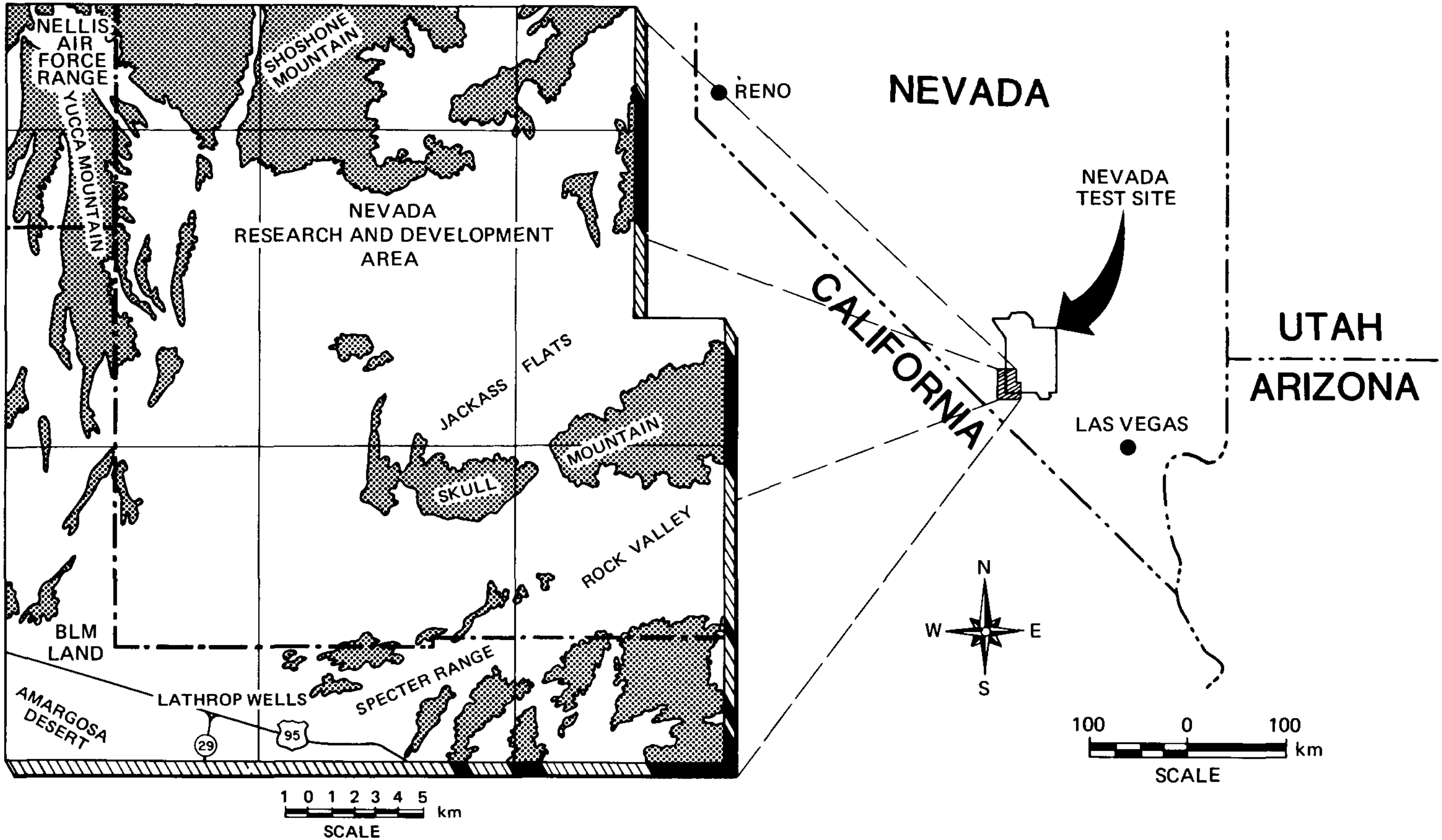

Figure 2 Nevada Nuclear Waste Storage Investigations' (NNWSI) repository location screening area.

Nature of the Screening Problem

Relation to the NWTS Site Characterization Process. Repository siting activities on the NTS are only one part of the comprehensive national effort to identify and characterize suitable repository sites and to develop engineered systems compatible with the site conditions. Sequential phases of the national site selection and characterization process, as outlined in the DOE's testimony before the Nuclear Regulatory Commission (NRC) (U. S. Department of Energy, 1980b and 1982), include:

1. National surveys to identify favorable regions (up to several states in extent) for repository development
2. Regional surveys to identify areas (up to 1000 square miles)
3. Area surveys, including limited drilling and field work, to identify locations (up to 30 square miles)
4. Location studies, including extensive drilling, testing, and field work; and conceptual repository design to identify specific sites (nominally 10 square miles)
5. Banking of a number of candidate repository sites
6. Concurrent detailed site studies including subsurface exploration, testing at the base of a large diameter shaft, and detailed facility design
7. Selection of a site or sites for application to the NRC for a license to construct a repository

The size of the NTS, which approximates that of an "area" in the NWTS site selection process, and the historical use of the NTS for nuclear activities prompted DOE to classify the NTS as an "area" not requiring identification by previous geographic screening (U. S. Department of Energy, 1980b and 1982). Accordingly the first screening step at the NTS, and the subject of this document, is to develop a method for evaluating the NTS area and for identifying locations for further study. A decision to proceed will initiate geologic and environmental studies of the location-specific phase of the NWTS site characterization process.

General Considerations. Several considerations affect the selection of a method suitable for screening the NTS (or any other area) for repository locations. First, the screening process must be able to objectively distinguish among alternative locations with respect to a set of multiple and commonly competing objectives for repository performance. A compatible set of usable, discriminating site selection criteria must be able to be derived from these objectives. Broad statements such as finding sites "compatible with waste containment, isolation, and retrieval" (U. S. Department of Energy, 1982) provide useful guidelines for location screening, but they cannot distinguish objectively among the relative merits of alternative locations. Therefore criteria are needed that specify what is meant by compatible, adversely affect, and other subjective statements of goals that appear in previously published general guidelines (U. S. Department of Energy, 1981a; International Atomic Energy Agency, 1977; National Academy of Sciences, 1978; Johnson and Deju, 1979; Cochran and Dimitri, 1979; Brunton and McClain, 1977; U. S. Nuclear Regulatory Commission, 1980). Because consensus on the content and importance of competing objectives and criteria will be difficult—if not impossible—to obtain, the method should include a means to evaluate the effects of criteria assumptions on screening results.

Second, existing information about the many physical factors considered important for repository siting must be organized in a consistent structure. Only then can criteria systematically be applied to an information base to identify more and less favorable locations.

Third, screening will be based on complex information characterized by varying degrees of uncertainty, disparity, and reliability. Data about the three-dimensional characteristics of the geologic, hydrologic, and environmental settings are available only for sparsely distributed locations throughout the screening area. Information about potential repository host rocks also varies with regard to the geographic distribution, the phenomenological responses, and how well the modeling characteristics defined in the laboratory transfer to in situ environments. Therefore many unavoidable judgments and assumptions will be contained in the information base. How well these judgments and assumptions are dealt with will be crucial to the screening results.

Finally, the screening method must be organized in a manner that allows interested parties, including regulatory agencies, to observe and assess the effects of assumptions, analysis logic, data uncertainties, professional opinions, and criteria definitions on the screening results. These basic considerations were paramount in the design of the screening method being applied at the NTS.

Performance Objectives

As expressed in the DOE's testimony to the NRC, the goal for radioactive waste disposal is "the effective isolation of radionuclides from the environment in a safe and environmentally acceptable manner" (U. S. Department of Energy, 1980b). Specific performance objectives by which a site will be judged by DOE before application for a construction permit are (U. S. Department of Energy, 1980b):

1. "Waste containment* within the immediate vicinity of initial placement should be virtually complete during the period when radiation and thermal output are dominated by fission product decay. Any loss of containment should be a gradual process which results in very small fractional waste inventory release rates extending over very long release times, i.e., catastrophic losses of containment should not occur."

2. "Disposal systems should provide reasonable assurance that wastes will be isolated† from the accessible environment for a period of at least 10,000 years with no prediction of significant decreases in isolation beyond that time."

3. "Risks during the operating phase of waste disposal systems should not be greater than those allowed for other nuclear fuel cycle facilities. Appropriate regulatory requirements established for other fuel cycle facilities of a like nature should be met."

4. "The environmental impacts associated with waste disposal systems should be mitigated to the extent reasonably achievable."

5. "The waste disposal system design and the analytical methods used to develop and demonstrate system effectiveness should be sufficiently conservative to compensate for residual design, operational, and long-term predictive uncertainties of potential importance to system effectiveness, and should provide reasonable assurance that regulatory standards will be met."

6. "Waste disposal systems selected for implementation should be based upon a level of technology that

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**Containment in this context refers to a capability for restricting the wastes within prescribed boundaries (e.g., the engineered waste package that will be emplaced in the floor or walls of underground tunnels).*

†Isolation in this context refers to the capability of the natural environment to prevent the migration of unacceptable amounts of waste to human populations.

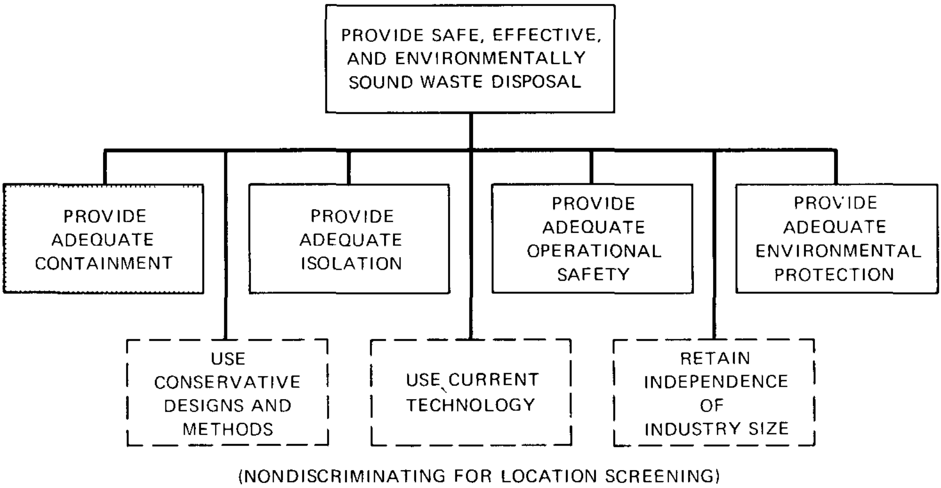


Figure 3 Major performance objectives for guiding NNWSI repository location screening. Lower levels of containment objectives (shading) are developed in Figure 4.

can be implemented within a reasonable period of time, should not depend upon scientific breakthroughs, should be able to be assessed with current capabilities, and should not require active maintenance or surveillance for unreasonable times into the future."

7. "Waste disposal concepts selected for implementation should be independent of the size of the nuclear industry and of the resolution of specific fuel cycle or reactor design issues and should be compatible with national policies."

Only the first four objectives are directly useful for distinguishing the relative merit of alternative geographic locations (Fig. 3). The last three are overriding considerations applicable to any and all potential sites and siting processes. By assuming certain limits for engineering flexibility, investigators can define desirable and undesirable conditions of the natural system with respect to each of the major location-distinguishing performance objectives. These conditions form the basis for a set of subobjectives around which the NNWSI screening method is structured so that all screening judgments and rationale are directly traceable to the four major objectives.

Objectives Tree. Repository performance objectives are organized into a hierarchical format referred to as an objectives tree (Fig. 4). The tree shows, from general to specific, how each of the four major performance objectives is to be achieved by searching for locations that optimize the chances for satisfactory performance with respect to individual subobjectives or criteria. An objectives tree is constructed by asking how each upper-level objective is to be accomplished (Fig. 4). The answer, or answers, must be comprehensive because they constitute a set of inclusive objectives of the next lower level. For example, the question "How can the DOE's objective one, adequate containment, be achieved?" can be answered by considering the things that may result in loss of containment and then by setting as sub-objectives

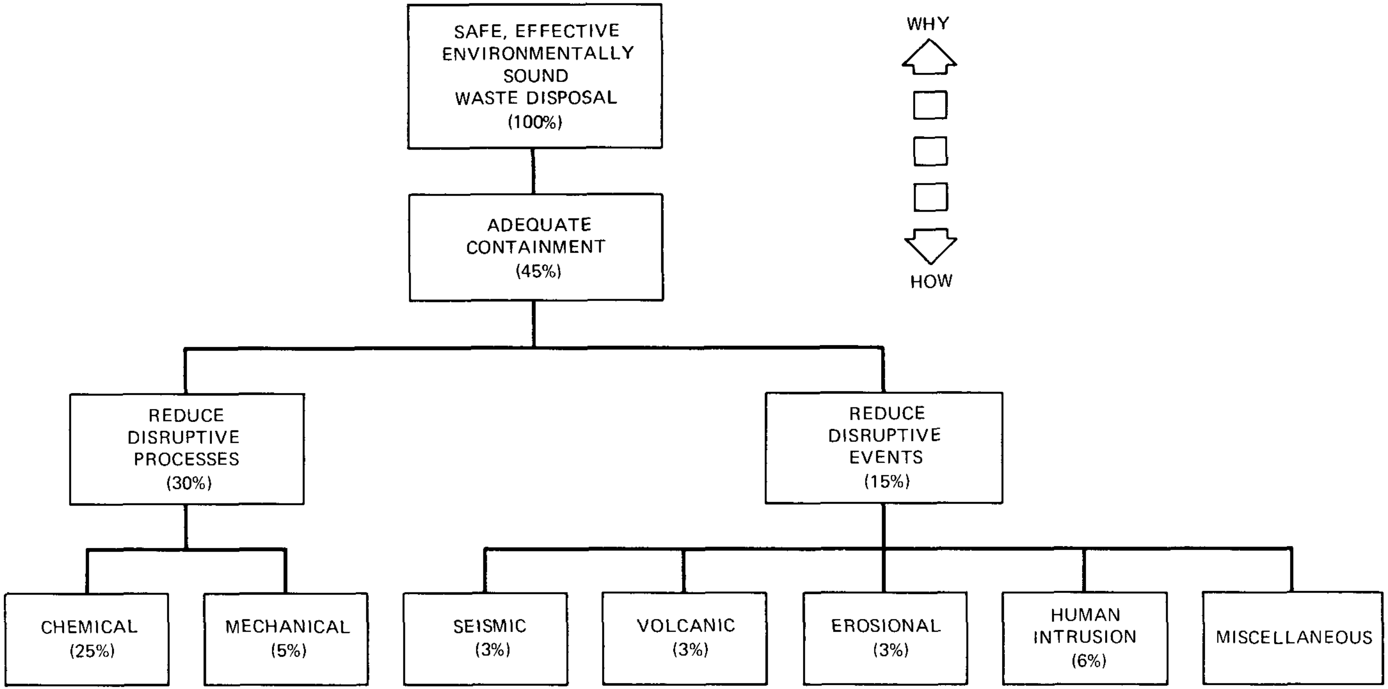


Figure 4 Example containment branch of an objectives tree. Percentages are hypothetical weighting factors discussed later under the heading "Rating of Alternatives."

the avoidance of those things. Loss of containment can result from disruptive processes that require long times to achieve equilibrium (relative to operational concerns) or from disruptive events that reach equilibrium in short times. Consequently sub-objectives for preserving containment include avoiding processes and events that would cause loss of containment. Together these two sub-objectives address all possibilities by which containment can be lost.

In turn, the questions "How can disruptive events be avoided?" and "How can disruptive processes be avoided?" form the basis for constructing the next lower level on the tree. Two processes, chemical and mechanical, address the processes by which containment can be lost and, therefore, constitute the sub-objectives for the process branch of the containment tree. Numerous types of events could cause loss of containment; however, seismic, volcanic, erosional, and human intrusional events are considered most credible. Therefore these events are the lower-level objectives of the event branch of the containment tree. A category of miscellaneous events was included in the tree to meet the requirement for a comprehensive listing of all possible answers to the question regarding how disruptive events could cause loss of containment.

The question, "Why are certain concerns necessary or important?" can be answered by inspection of an objectives tree. Each higher-level objective provides the rationale for pursuing its set of lower-level objectives.

Readers should note that the lower-level objectives could be divided into sets of sub-objectives. Seismic events, for example, could be separated into fault movements that may shear waste packages and vibratory ground motion that may cause failure of the packages. We stopped developing the tree at the second level below each of the four major DOE performance objectives because, given the state of knowledge about the NTS and its surroundings, that level of development was judged sufficient to resolve the problem into components compatible with evaluation.

Readers also should note that there is no unique solution to the problem of separating major objectives into sets of hierarchical sub-objectives. Although different approaches to organizing an objectives tree are possible, if differently organized trees are comprehensive of concerns relevant to a given problem, the trees will converge at their lower levels on the information required to evaluate the problem. The lowest level sub-objectives of the NNWSI screening are consistent with requirements for the natural system outlined in *NWTS Program Criteria for Mined Geologic Disposal of Nuclear Waste: Site Performance Criteria* (U. S. Department of Energy, 1981a) and with other published repository siting criteria documents (International Atomic Energy Agency, 1977; National Academy of Sciences, 1978; Johnson and Deju, 1979; Cochran and Dimitri, 1979; Brunton and McClain, 1977; U. S. Nuclear Regulatory Commission, 1980).

Physical Attributes

Repository Model. To meet its objectives, particularly NWTS objective six, "use of current technology," DOE has chosen to concentrate nuclear waste management efforts on one particular disposal concept, mined geologic repositories (U. S. Department of Energy, 1980a, 1980b, and 1981b; Presidential Message to Congress, 1980). Accordingly NNWSI siting activities currently are directed toward identifying locations compatible with the characteristics of a mined

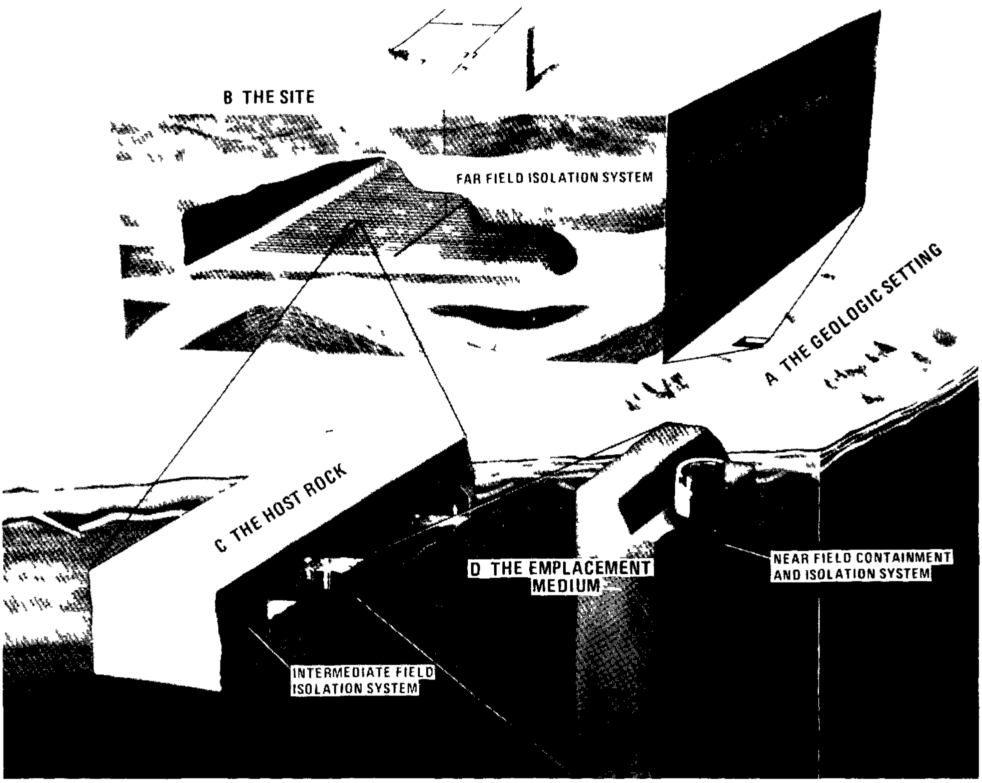


Figure 5 General model of a mined geologic repository.

repository. This concept calls for emplacing specially packaged, solid radioactive waste forms in holes drilled into the walls or floors of tunnels hundreds of meters below the ground's surface (Fig. 5).

To enhance confidence that a mined repository will perform as intended, investigators have separated the repository concept into components. Each of these components can be independently and impartially assessed in terms of its contribution to performance with respect to each element of the objectives tree. The most general of these components are the engineered system and the natural system. The engineered system is composed of the waste form, its package, the subsurface excavations, the waste emplacement design, the waste transport mechanisms, and the waste handling facilities that are at the earth's surface and within the excavations. Details of the engineered system cannot be determined until a site has been selected. Therefore consideration of alternative engineering concepts must be postponed until a specific location is selected and investigated. For this reason, only natural conditions of the NTS and its surroundings will be the basis for location screening.

The natural system is composed of geologic, hydrologic, meteorologic, and ecologic systems. These systems generally are beyond engineering control and therefore must be selected, rather than designed, for properties that would inhibit mobilization, subsurface transport, and surface dispersal of radioactive contaminants should the engineered components fail. Properties of the natural system that allow an evaluation of the degree to which portions of the NTS screening area satisfy the lower-level objectives on the tree have been organized according to a hierarchy of topical categories. These categories range from the far-field general setting of a site to the very near-field waste emplacement medium. This hierarchy is illustrated in Fig. 6.

Attribute Maps. These topical categories of natural features, referred to as attributes, must be measurable and mappable conditions, or properties of the natural system, and they must provide information suitable for evaluating performance with respect to one or more lower-level objectives on the objectives tree.

For instance, one lower-level objective is to reduce hazards associated with volcanic activity; relevant and measurable parameters of the natural system for evaluating that objective might be the distance from the most recently active volcanic belts or the likelihood of volcanism as a function of local structural conditions. A separate map will be compiled for each attribute that is determined to be suitable for discriminating among locations in the southwest NTS. Selection of appropriate measuring parameters for the attributes and for development of the maps will be the responsibility of appropriate experts within the NNWSI project and others familiar with the attributes in question. Considerable professional judgment will be required in both these endeavors.

Because much information from current repository exploration and characterization studies at the NTS is preliminary, sparsely distributed, or available only for isolated portions of the study area, it is unavoidable that investigators will have varying levels of confidence in the mapping data. Confidence in the supporting information usually has not been treated in a systematic manner in previous site screening analyses. This causes concern because alternative screening locations may appear similar with respect to overall suitability, but the ratings may be based on information sources of varying reliability. We believe that systematic consideration of such confidence differences is important to a decision on repository siting. Therefore we hope to prepare maps analogous to the reliability diagrams accompanying many AMS and USGS 1° × 2° topographic quadrangles. These maps would express judgments regarding confidence in the mapped data. If we apply this option, the same technical personnel who will prepare the maps of attribute measures will provide the confidence estimates. Techniques for evaluating the reliability of the screening information and for mapping the various levels of confidence will be developed after the attribute maps are completed and assessed.

Relation of Attributes to Objectives. A matrix that relates performance objectives to screening attributes (Fig. 7) establishes the applicability of each attribute to evaluating the suitability of each location with respect to individual lower-level objectives. This matrix systematically resolves the large, complex screening problem into smaller, more manageable subsets. The matrix also provides a convenient framework for documenting the screening analysis in a logical and comprehensive fashion.

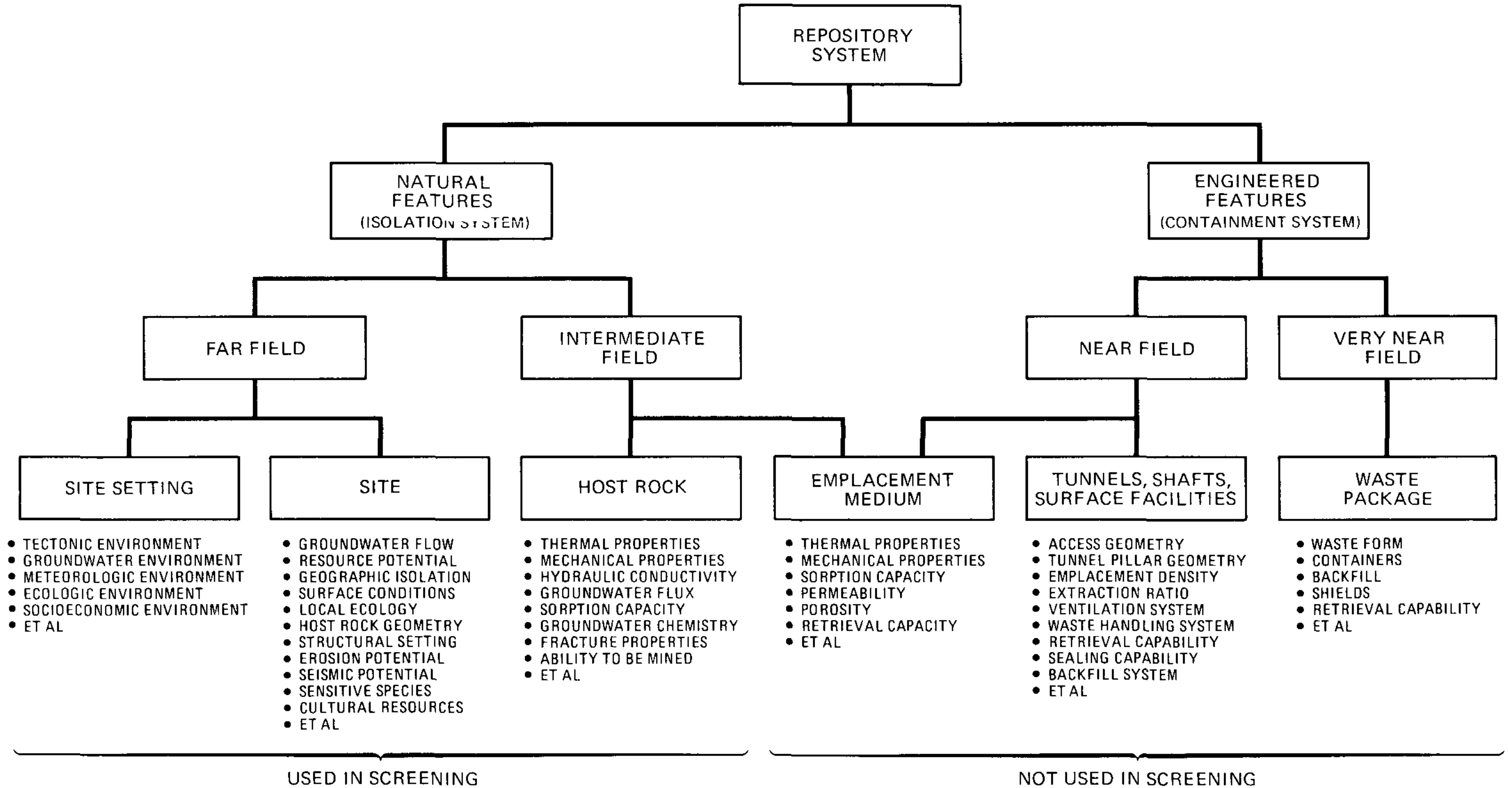


Figure 6 Hierarchy of repository attributes. Note that the emplacement medium is distinguished from the host rock to accentuate the scale at which the engineered and natural systems converge.

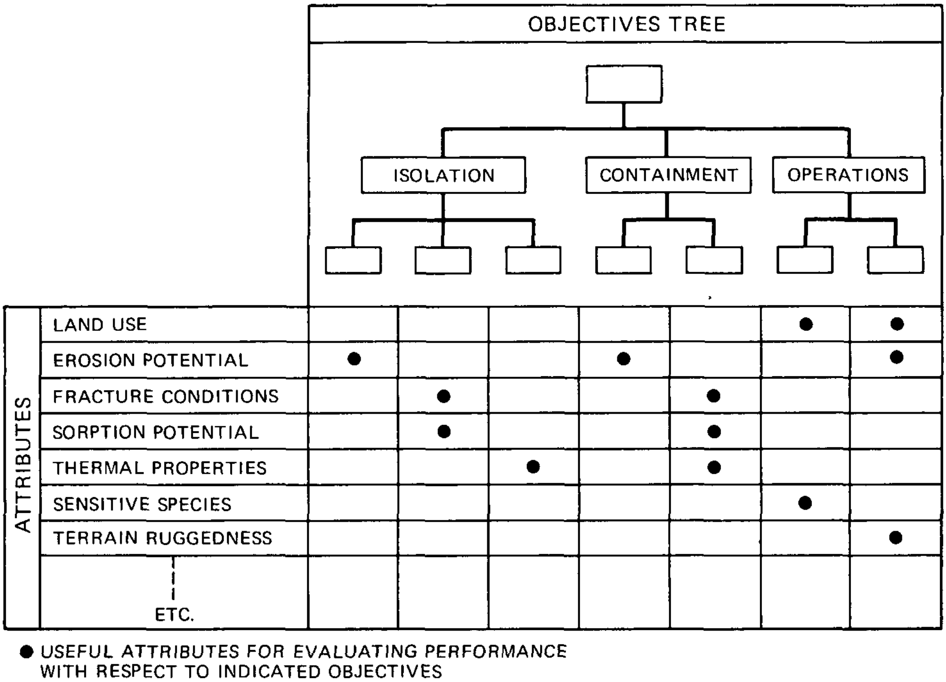


Figure 7 Schematic matrix illustrating how physical attributes of the natural system relate to evaluating one or more performance objectives.

Mapping parameters for the attributes will be specially selected to organize natural-system data within a context appropriate for evaluating the objectives. These mapping parameters will not include judgments of favorability. For example, thermal conductivity of a potential host rock is a consideration for evaluating a repository's mechanical and chemical performance; it may be measured and mapped. However, a map of thermal conductivity values expresses physical facts about the screening area; it does not convey information about whether the physical fact is good or bad for repository performance. That assessment requires a separate judgment.

Favorability Functions

Favorability functions provide the required links between factual conditions of the physical world and the desired conditions for repository performance. Such functions will be developed in a manner that ties conditions of the mapped attributes to desires expressed by lower-level objectives. These functions are a form of siting criteria in that the objectives define a set of desirable goals to be pursued; the attribute maps express how physical conditions of the natural system are distributed throughout the screening area; and the favorability functions indicate the degree to which the physical conditions are compatible with the goals.

A separate favorability function will be developed for each attribute. Each function can be expressed as a graph. Units on the abscissa must be identical to mapping units for the attributes; the ordinate is a standardized scale of favorability with regard to a particular performance objective. For this application the scale will be defined by values ranging from 0 (relatively unfavorable) to 10 (relatively favorable) (Fig. 8).

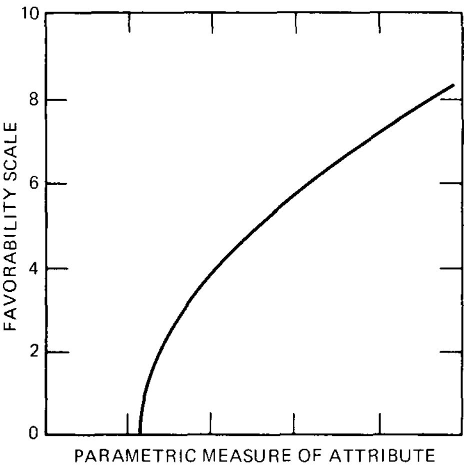


Figure 8 General features of a favorability function relating measures of an attribute to a scaled favorability rating.

A single attribute may apply to different objectives in different ways and may require more than one favorability function. Porosity, an attribute of the geohydrologic system, may in some cases be undesirable with respect to groundwater flow time for some rock types, but it may be desirable with respect to sorptive capacity.

Thus this method of analyzing the physical conditions allows judgments about natural system favorability to be made with regard to separate objectives independent from judgments about the physical conditions of nature as represented on the attribute maps. Favorability functions also allow the systematic consideration of more favorable vs. less favorable conditions. Consequently, as opposed to evaluations based solely on a set of exclusionary conditions, favorability functions increase the information upon which decisions are made. If appropriate, threshold levels can be included in the favorability functions to permit exclusionary conditions of the natural environment to be considered.

Assigning relative favorability values to attribute properties for each appropriate objective will require considerable insight and judgment on the part of the experts and managers of the NNWSI project. Although general trends are expected to be commonly agreed on, consensus on the exact favorability values for individual functions probably will be elusive.

Because the predicted ultimate radiological dose-to-man health impact is the primary criterion that constrains long-term suitability of a repository, any design or siting criteria for assurance of radiological safety that are expressed in terms of other factors are necessarily arbitrary. Ultimately, unavoidable judgments about the nature and geographic distribution of parameters of the mapped attribute conditions and judgments about the relations of attributes to siting objectives must guide site screening and site selection activities. The NNWSI screening method is designed to apply a systematic rationale to as many of these judgments as possible. Each earth science factor used to represent natural conditions considered in site screening will be reduced wherever possible to quantitative expressions. Without a full safety assessment that couples all of these quantitative expressions into a set of radiological dose-to-man predictive models, the presumed relations of these quantitative expressions for attribute conditions to the ultimate repository performance (as described by the favorability functions) must serve as the basis for location recommendation. This emphasizes the difficulty in specifying criteria and exercising objective site selection prerogatives before the full nature and the impacts of all site conditions are known.

Mapped qualitative expressions about existing natural conditions and the likelihood of hypothetical disruptive events will help to standardize the attribute information base and to guide the judgments of persons responsible for decisions regarding site suitability. However, the possibility that the decisions may turn out to be wrong persists, and the only alternative is to delay location suitability judgments until enough data for each potential site are collected to allow the full capabilities of a safety assessment to be exercised. Given that many potential locations exist and that unavoidable quantitative uncertainties will occur even in the most complete safety assessments, this alternative is unrealistic and counterproductive to the national effort to site and construct a repository. Therefore the appropriate question is not "How can we guarantee, before full data collection, the selection of a safe site?", but "How can we standardize the incomplete information base for our judgmental selection process?"

The criteria (favorability functions) and data reduction methods discussed in this report are means used by the NNWSI before completion of a full safety assessment for guiding the identification of geological systems that are judged, based on current information, capable of ultimately providing acceptable repositories. Because this phase of the NNWSI siting activity is a geographic screening designed to identify suitable locations for focusing exploration resources and is not a safety assessment (a safety assessment will not be completed until after the data from subsequent characterization studies are available), no rigorous, comprehensive effort will be made at this time to define favorability scales for all attributes in terms of rigid acceptability criteria. However, the method outlined here is adaptable to application of absolute quantitative criteria should consequence assessments or other analytical means become available to allow an absolute rating of physical conditions for repository performance.

Evaluation of Alternative Locations

Definition of Alternative Screening Locations. Alternative geographic candidates for screening can be defined in a number of ways. One is by dividing the southwest NTS into discrete geographic units based on potential host rock continuity at the depths of interest. Another is by designating locations with similar physiographic (e.g., Skull Mountain, Jackass Flats, etc.) or geohydrologic characteristics. A more abstract technique involves dividing the southwest NTS into a large number (a few thousand) of resolution units arranged upon an arbitrarily imposed geographic grid, such as quarter sections (½ mile by ½ mile). By this technique individual attribute maps are overlaid on the grid (Fig. 9). The size of each grid unit is considerably smaller, perhaps by a factor of 10 to 30, than the area required for a repository. Each grid unit can be independently analyzed for suitability; and repository locations can be defined where an appropriate number of contiguous units with favorable ratings occur. This technique therefore both identifies and evaluates alternative locations. As a result, the likelihood is reduced for bias from defining alternatives on a priori notions about location boundaries.

Many concerns about repository performance depend on the properties of the host rock in which wastes will be emplaced. Therefore the NNWSI will combine a geographic grid with occurrences of host rock to define alternative locations. Each potential

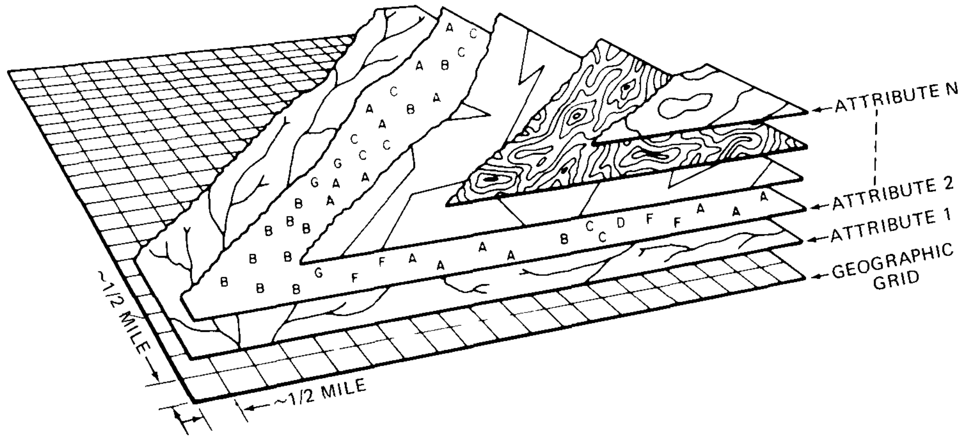


Figure 9 Schematic overlay of individual attributes on an arbitrary geographic grid encompassing a screening area.

host rock in the southwest NTS will be delineated where it occurs at appropriate depths and with sufficient thickness. A quarter-section grid also will be constructed for the entire southwest NTS screening area.

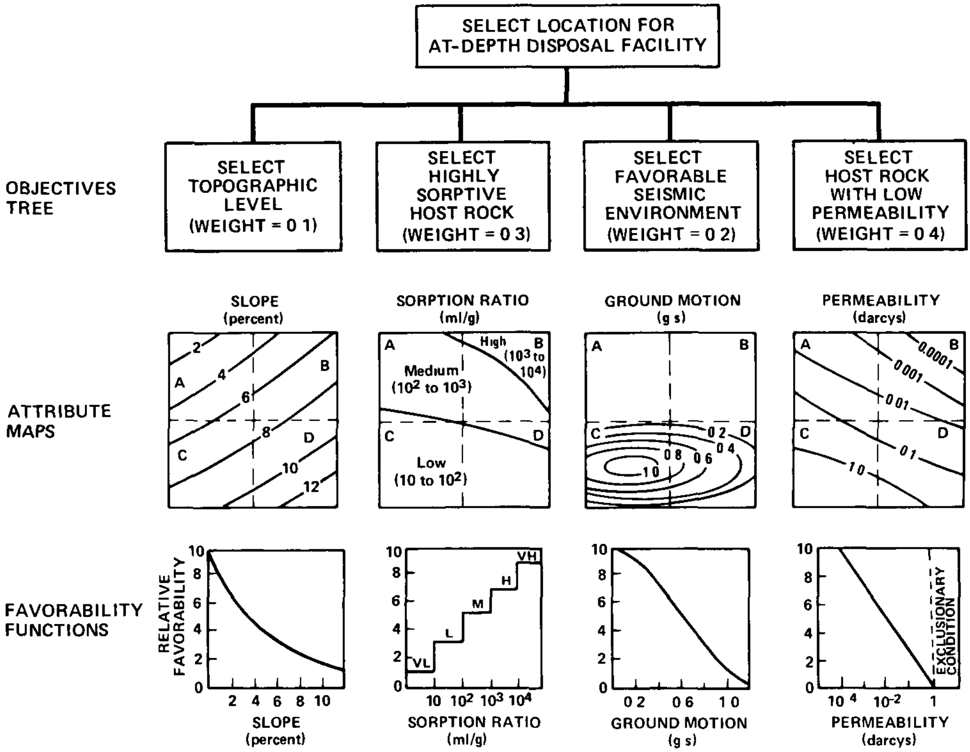
Attribute properties that depend on rock type will be assigned to grid elements corresponding to geographic occurrences of the appropriate rocks. Properties that vary independently of rock type, notably surface properties such as occurrences of sensitive species and terrain factors, will be assigned to grid elements throughout the screening area. In this manner multiple grid locations corresponding to different host rocks at different depths can be evaluated at a single geographic position. Each grid element in the southwest NTS will be independently analyzed even though some of the grid elements will not possess a designated host rock at depth, whereas others will possess more than one. This hybrid technique for alternative definition permits consideration of the merits of locations that lack only a host rock that is presumed satisfactory. Therefore a location that appears exceptionally favorable, except that it does not possess a presumably satisfactory host rock, may prompt a legitimate reconsideration of what constitutes acceptable host rock properties.

Rating of Alternatives. Evaluation of individual grid units will be performed by a set of simple computer algorithms. The procedure by which performance rating scores can be obtained for each geographic grid element can be generalized as:

1. Determine the physical value of each attribute for each grid element from the attribute maps
2. Determine the favorability number for that attribute value by comparing the results of step one to the favorability functions
3. Sum the individual favorability numbers for each geographic grid element according to the assigned weighting factors (discussed below)

Grid elements with higher total scores are more favorable. This process is illustrated in Fig. 10, which shows the way rating scores are computed for a hypothetical screening area composed of four geographic grid elements, A, B, C, and D.

An important element of the screening method, the weighting of



COMPUTATION OF RATING SCORES FOR ALTERNATIVE LOCATIONS

		ATTRIBUTES				
		SLOPE	SORPTION RATIO	GROUND MOTION	PERMEABILITY	
STEP 1	ALTERNATIVE LOCATIONS	AVERAGE ATTRIBUTE VALUES OF ALTERNATIVE LOCATIONS (FROM ATTRIBUTE MAPS)				
	A	4	M	0 1	0 01	
	B	6	M-H	0 1	0 001	
	C	8	L	1 0	1 0	
	D	10	L	0 5	0 1	
STEP 2		FAVORABILITY VALUES (F) FOR INDIVIDUAL ATTRIBUTES (FROM GRAPHS AND STEP 1)				
	A	4 5	5	9 5	5 0	
	B	3 0	6	9 5	7 5	
	C	2 0	3	1 0	0★	
	D	1 5	3	0 5	2 5	
STEP 3		CONTRIBUTION OF EACH ATTRIBUTE TO OVERALL RATING [WEIGHT (W) FROM OBJECTIVES TREE TIMES FAVORABILITY (F) FROM STEP 2]				TOTAL RATING
		0 1 x F	0 3 x F	0 2 x F	0 4 x F	Σ (W x F)
	A	0 45	1 5	1 9	2 0	5 9
	B	0 30	1 8	1 9	3 0	7 0
	C★	0 20	0 9	0 2	0★	1 3
	D	0 15	0 9	1 2	1 0	3 3

★ ALTERNATIVE LOCATION C IS EXCLUDED FROM FURTHER CONSIDERATION BECAUSE ONE ATTRIBUTE (PERMEABILITY) POSSESSES AN EXCLUSIONARY CONDITION

Figure 10 Example analysis showing the way components of the NNWSI screening method are combined to obtain a rating score for each grid unit (A, B, C, and D).

various performance objectives and attributes, is illustrated on Figs. 4 and 10. In any analysis with multiple competing objectives, such as repository siting, some objectives are considered more important than others. For example, should equal importance be given to desires to find sites with long groundwater flow times to the biosphere and to find sites with minimal potential for meteorite impacts? In this extreme example the answer obviously is no; long flow paths are unequivocally more important. In fact, all separate objectives may be presumed to have a different relative importance to the overall goal of safe, environmentally sound, cost effective waste disposal. A system of weighting will be used to account for the differing importance of individual objectives. If the assumption is made that the overall problem of finding a satisfactory site has an importance of one hundred percent, then each lower-level sub-objective will have an importance that is a lower percentage. The sum of the weights for all the sub-objectives of the same level will be one hundred percent.

The hierarchical nature of the objectives tree facilitates assigning relative importance to each level of sub-objectives. The tree allows weighting to be approached by a series of iterations that progress from general to specific, and it alleviates the problem of attempting to determine the relative weight of all lower-level objectives in one step.

The favorability value of each attribute (Fig. 10, step 2) will be multiplied by the weight of the objective that is addressed by that attribute (Fig. 10, top row). The sum of the weighted favorability values thus forms the basis for rating the alternative grid locations.

As with construction of favorability functions, determination of weighting values for objectives will require considerable insight and judgment. Management personnel as well as technical experts are required for this task. This is especially true for considering trade-offs among higher-level objectives, such as safety, environment, and construction cost.

Sensitivity Analyses. Many judgments are required to support this screening method. First, expert opinions about the physical conditions in the screening area and about the confidence that can be placed in those opinions must be recorded on attribute maps. Second, judgments are required about relative favorability (on a scale of 0 to 10) with regard to the various physical conditions mapped for each attribute. Finally, judgments are required about the relative importance of various performance objectives. Each of these judgments will affect the numerical results of the screening analysis; and it is highly unlikely that a satisfactory consensus can be achieved for each of the judgments to be made. Nonetheless, objective means are not available to alleviate the need for such judgments whether screening is performed by the method outlined in this paper or by any other method.

To account for these subjective elements, investigators will evaluate the sensitivity of the screening results for a range of reasonable judgments concerning the geographic distribution of attribute properties, the shape of favorability functions, and the relative importance of the performance objectives. Each of these parameters will be varied within reasonable limits and the effects on the ratings of alternative locations will be assessed.

The screening activity will not produce a unique solution; rather, the method will provide a decision-support base with a range of options defined by assumptions about attribute data, relative weights of performance objectives, and favorability functions. Policy makers will have the responsibility to determine which set of assumptions to follow. Once such decisions have been made, the screening activity will highlight locations having the highest overall potential for suitability for a repository and, therefore, the best locations for further exploration and characterization efforts. Furthermore, by assigning zero weights to some objectives, investigators can use the screening method to investigate which locations are most suitable based on a limited set of specific attributes or objectives.

Documentation

Each component activity of the screening process will be documented. Documentation will include sources of information, assumptions, analytic logic and personnel used to construct the objectives tree, attribute maps, and favorability functions. Information will be recorded on individual data sheets for each objective, map, and function. Data sheets will be organized and filed for easy retrieval. Computer algorithms used to digitize the data base and to perform numerical evaluation of the screening area also will be documented. Different sets of assumptions used in sensitivity analyses will be recorded, and location ratings associated with each set of assumptions will be preserved and filed. Based on the results of the sensitivity analysis, the NNWSI Technical Overview Contractor will summarize in a Screening Summary and Recommendation Report recommendations concerning the locations, if any, that should be explored further. This report will be transmitted to the NNWSI Site Evaluation Working Group.

Acknowledgment

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Status of Gulf Coast Salt Dome Characterization

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Screening and characterization for a potential nuclear waste repository have progressed through the area phase in these Gulf Coast Salt Basins. The domes studied during the area phase are described briefly. The area characterization studies are outlined, and the resulting reports are listed. Geologic and environmental studies resulted in elimination of four domes from further consideration. The remaining domes were judged acceptable and were classified as to their favorability to license.

Site characterization planning for location phase activities deals primarily with technical, environmental, and socioeconomic issues of concern to the states and/or to the Office of Nuclear Waste Isolation (ONWI), Department of Energy (DOE). These issues are listed and discussed. 16 ref.; 9 fig.

Introduction

Characterization of Gulf Coast salt domes has progressed from regional characterization and screening of the inland and off-shore salt basins of the Gulf Coast through area characterization and screening of eight domes located in three salt basins in Texas, Louisiana, and Mississippi. Geologic/hydrologic and environmental/socioeconomic characterization reports for both the regional and area phases have been produced and distributed to the states. The Office of Nuclear Waste Isolation (ONWI) has recommended four domes for further studies in the location phase. Planning for the characterization and screening of these four domes is directed by ONWI and is coordinated by ONWI's Geologic Project Manager (GPM) and Regulatory Project Manager (RPM). Participants include state and federal agencies and state representatives.

Origin of Salt Domes

A salt dome is a diapiric structure—often miles in radius—that may extend from depths of 15,000 to 30,000 ft up through younger sediments to within a few feet of the surface. Most domes, however, are overlain with several hundreds to thousands of feet of sedimentary rock.

Salt domes in the United States occur in the Gulf of Mexico, along the coastal areas of the Gulf, and in interior basins. Gulf domes are thought to be extremely young and active because the Gulf presently is undergoing sedimentation. Coastal domes are thought to be much older and inactive. Both areas containing domes are noted for hydrocarbon production. The salt domes of the three interior basins, one each in northern Louisiana, eastern Texas, and southern Mississippi, are not considered active and have limited hydrocarbon potential. The tops of many domes are covered by a caprock—generally anhydrite, gypsum, calcite, and/or limestone.

The formation of the caprock and the mechanism of salt dome formation are subjects of much debate. Geologists generally agree on three points concerning the origin of salt structures:

- Salt domes have been derived from a sedimentary salt bed.
- Plastic flow has created diapiric structures (domes).
- Density differences between the salt and the overlying sediments plus down-building of sediments because of loading around the domes are sufficient to cause dome development.

The Louann Salt of Late Triassic-Early Jurassic age is the source of mother salt for dome formation. Presumably the salt was deposited in an evaporating sea to a thickness of several thousand feet. From the Late Jurassic through the Tertiary, a time span of 120 million years, virtually continuous sedimentation buried the salt layer to a depth of several miles. This sedimentary loading produced both lateral and vertical plastic movement of the salt; the initial mobility possibly, but not necessarily, was associated with tectonic activity. Initial salt movement produced pillows and ridges. During the Cretaceous Period, diapirs developed from the deeper pillows to form the salt domes.

Because the mobile salt formed a substratum for overlying sedimentation, the salt movement affected the thickness and lithology of the overlying formations. Areas that the salt evacuated to form these growing structures became structural basins and structural depressions, referred to as rim synclines, that partially or completely encircle most domes. Such depressions are thought to be caused by the overlying sediment sinking into the space left during the lateral flow of the salt to the dome. Rim synclines are thought to typically migrate inward and upward with time as the salt in the mother bed is depleted.

Salt dome interiors are almost pure sodium chloride. The salt is coarse-grained and is distinctly crystalline with prominent cubic cleavage. Most halite crystals range from 0.25 to 0.5 in. in diameter. However, fine- to very-fine-grained salt also occurs. Pods of extremely coarse-grained salt—usually aligned parallel to the banding—with crystals one to two inches or more across also occur.

Minor amounts of anhydrite, commonly one to ten percent, are present in the salt either as finely disseminated crystals or as large fragments. A small amount of anhydrite results in a dark discoloration contrasting to white salt. This produces banding representing original depositional layers.

Carbonates, sulfides, and sulfate minerals also have been reported. Minor amounts of potash appear in some domal salt, and inclusions of country rock, such as sandstone, occur in some domes.

Porosity and permeability of domal salt vary from low to virtually nonexistent. Inclusions of brine, oil, and gas are in some domes, and minor hydrocarbon seeps have been found in some salt mines, particularly those located near or on the Gulf Coast.

Caprock composed of anhydrite, gypsum, and/or limestone occurs on all the domes studied. Generally the limestone is at the top of the caprock, and anhydrite is at the base; gypsum and some calcite are in the middle. Anhydrite also may drape down the sides like a hood. Additionally, anhydrite forms on and over edges of overhangs and, in some cases, extends beneath overhangs. In most cases the caprock is thickest and most extensive over dome centers. Normally the caprock is 200 to 400 ft thick; however, it can be absent, or it can be as much as 1000 ft thick (Piece and Rich, 1962).

Current views on the mechanism of caprock formation support either the residual accumulation theory or the precipitation-in-place theory. The residual accumulation theory assumes that the caprock was formed at the top of the salt dome some time in the past as the salt was dissolved by groundwater and the less soluble materials (mainly anhydrite) were left behind. Such anhydrite constitutes a small percentage of the original salt mass and generally is believed to have been deposited with the salt from seawater when the Louann Salt beds were formed. An alternate explanation by Paulson (1977) suggests that some of the caprock has been brought up along with the developing dome from evaporites originally beneath the salt bed in the evaporite basin. The second theory, precipitation in place, assumes that salt brine from deep saline aquifers rose along the salt dome stock and precipitated caprock on top of the dome when the dome top came into contact with a fresh-water aquifer.

Dome Descriptions

Each of the eight domes identified in the region-to-area screening process was investigated during the area characterization phase. Results of studies conducted by the GPM are summarized below in brief descriptions of each dome.

Rayburn's Dome. Rayburn's Dome (Fig. 1) is located in southeastern Bienville Parish, Louisiana, six miles northeast of the town of Saline and two miles west of the community of Friendship. The major portion of the land is farmed. Rayburn's Dome is expressed at the surface by a circular topographic low surrounded by hills with approximately 100 ft of relief. A saline, or salt lick, occupies the central depression into which intermittent streams from the surrounding hills flow. The saline is drained to the south by Fouse Creek, which is a tributary of the Dugdemona River.

The top of the caprock in wells in Rayburn's Dome is believed to range from +179 to +65 ft above Mean Sea Level (MSL); it covers the top of the salt and drapes the flanks according to gravity modeling. Caprock thickness varies from 30 to 88 ft. The top of the salt in wells at Rayburn's Dome ranges from +90 to −23 ft MSL.

Rayburn's Dome is pear-shaped in plan view with the long axis oriented northwest-southeast. Estimated horizontal area is 790 acres at −2000 ft MSL, 924 acres at −2500 ft MSL, and 1082 acres at −3000 ft MSL.

Vacherie Dome. Vacherie Dome (Fig. 2) is located in southeastern Webster and northwestern Bienville Parishes, Louisiana, two miles east of the town of Heflin. Most of the land in the dome area is in forest; a small portion of the area is used for farming. Vacherie Dome is expressed at the surface by a northwest-southeast oriented topographic low surrounded by hills with approximately 140 ft of relief. Bashaway Creek drains the central depression to the east into Black Lake Bayou, which is a tributary of the Red River.

Elevations of the top of the caprock in wells in Vacherie Dome range from −303 to −1331 ft MSL. By using gravity modeling and well data, investigators have interpreted the caprock to cover the top of the salt and to drape over the flanks. Caprock thickness varies from 79 to 273 ft. Elevations of the top of salt in wells at Vacherie Dome range from −560 to −1410 ft MSL.

This dome is elliptical in plan view and ridge-like in profile; the long axis trends northwest-southeast. Estimated horizontal area is 2031 acres at −2000 ft MSL, 2379 acres at −2500 ft MSL, and 2730 acres at −3000 ft MSL.

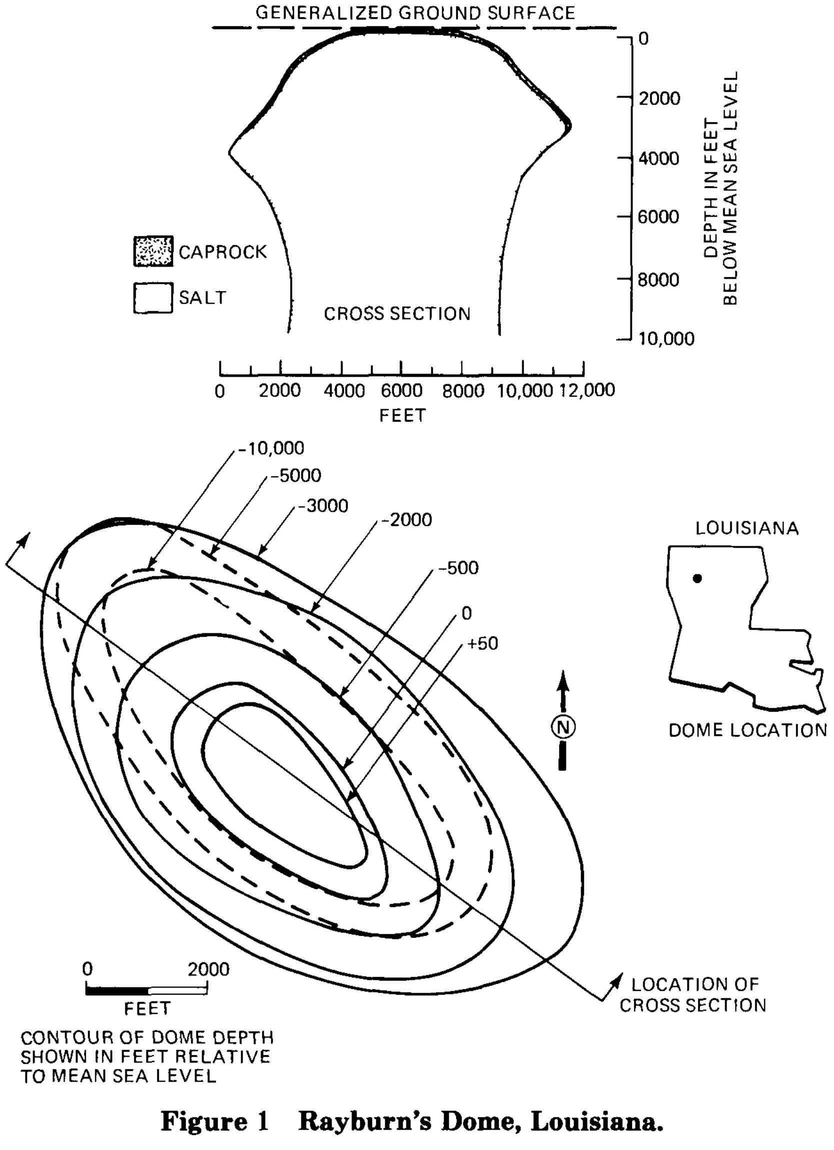

Figure 1 Rayburn's Dome, Louisiana.

Cypress Creek Dome. Cypress Creek Dome (Fig. 3) is located in south central Perry County, Mississippi, four miles southeast of the town of New Augusta; the dome is within the boundaries of the Camp Shelby military facility and the DeSoto National Forest. Most of the land in the dome area is managed forestland. The dome lies beneath a broad swamp at the headwaters of Cypress Creek. Cypress Creek is a tributary of Black Creek which flows southeasterly into the Pascagoula River. The topographic depression over the dome may be its surface expression. Relief in the dome area is approximately 80 ft.

Evaluations of the top of the caprock in wells at Cypress Creek Dome range from −951 to −1120 ft MSL. Caprock thickness varies from 35 to 204 ft. Elevations of the top of

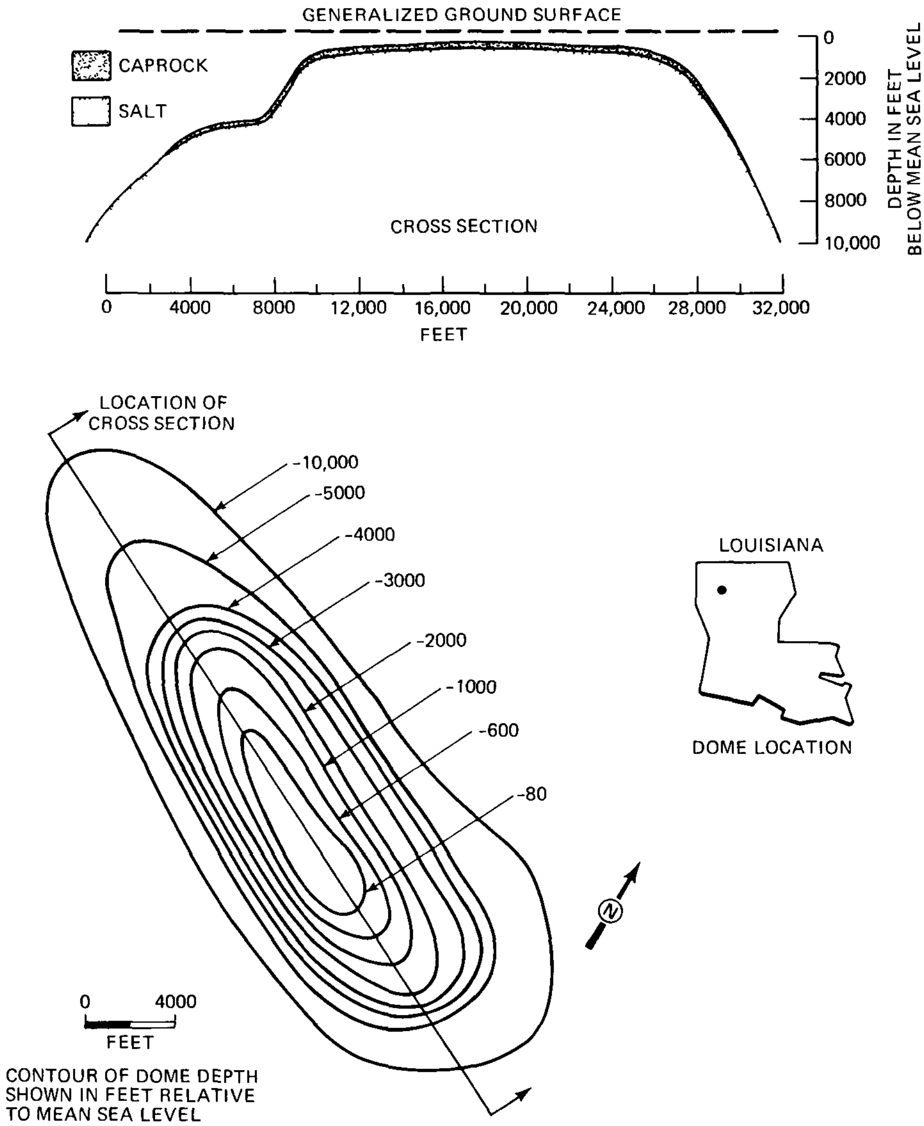

Figure 2 Vacherie Dome, Louisiana.

the salt in wells range from −1037 to −1184 ft MSL.

Cypress Creek Dome is roughly elliptical in plan view with the long axis oriented northeast-southwest. A circumferential salt overhang that ranges from −2500 to −6000 ft MSL has been inferred on the basis of gravity modeling and well data. Six petroleum exploration wells, all of which penetrated caprock and salt, have been drilled through the overhang. Three of the wells are actively producing oil and gas. The estimated horizontal area at Cypress Creek Dome is 2598 acres at −2000 ft MSL, 2416 acres at −2500 ft MSL, and 2249 acres at −3000 ft MSL.

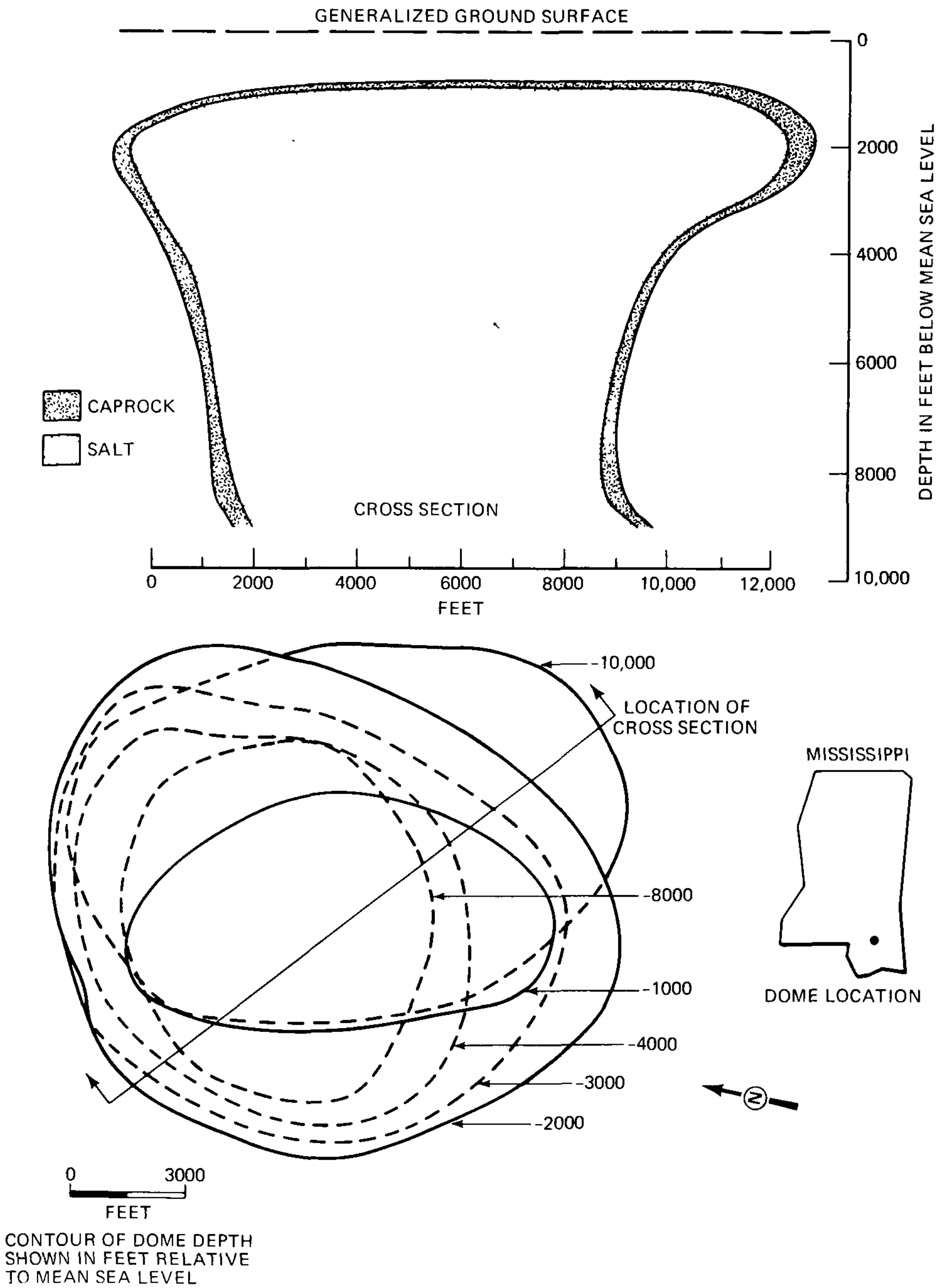

Figure 3 Cypress Creek Dome, Mississippi.

Lampton Dome. Lampton Dome (Fig. 4) is located in east central Marion County, Mississippi, six miles southeast of the city of Columbia and 4.5 miles east of the town of Lampton. Most of the land in the dome area is managed forestland that includes a game reserve; the remaining land is used for farming or livestock grazing. The dome is located beneath an east-west trending, flat-topped ridge that divides drainage between Upper Little Creek and Lower Little Creek, which are tributaries of the Pearl River to the south. Topographic relief in the dome area is approximately 180 ft.

Elevations of the top of the caprock in wells at Lampton Dome range from −1000 to −1293 ft MSL. Caprock thickness varies from 37 to 262 ft. Elevation of the top of salt, encountered in only one well, at Lampton Dome is −1360 ft MSL.

Lampton Dome is elliptical in plan view with the long axis oriented northwest-southeast. A salt overhang, which circumscribes the dome between the −8000 and −16,000 ft MSL levels, has been interpreted on the basis of gravity modeling.

Richton Dome. Richton Dome (Fig. 5) is located in northern Perry County, Mississippi. The small town of Richton is located approximately two miles east of the center of the dome. Most of the land in the vicinity is managed forestland; a small portion of the land is used for farming and livestock grazing. Richton Dome is characterized by lithologically controlled topography with approximately 150 ft of relief. The dome is located beneath the drainage divide between Bogue Homo and Thompson Creek, which are tributaries of the Leaf River to the south.

Elevations of the top of the caprock in wells at Richton Dome range from −264 to −654 ft MSL. Caprock varies in thickness from 20 to 213 ft. Elevations of the top of salt in wells range from −508 to −593 ft MSL.

Richton Dome is elliptical in plan view and ridge-like in profile; the long axis is oriented northwest-southeast. Well data indicate an extensive overhang on the western flank and on the northeastern flank of the dome. Estimated horizontal area is 5585 acres at −2000 ft MSL, 5847 acres at −2500 ft MSL, and 5825 acres at −3000 ft MSL.

Keechi Dome. Keechi Dome (Fig. 6) is located in north central Anderson County, Texas, five miles northwest of the city of Palestine and 2.5 miles southeast of the community of Montalba. About half of the land in the dome vicinity is in forest and half is pasture. The topographically low area over the dome is surrounded by a ring of hills with relief of up to 140 ft. Keechi Creek and Six Mile Creek drain the central area of the dome and flow southward into the Trinity River.

Elevations of the top of caprock in wells at Keechi Dome range from +225 to −3066 ft MSL. Caprock is believed to drape the dome and extend down the flanks; its thickness varies from 16 to 310 ft. A caprock overhang has been interpreted at Keechi Dome on the basis of gravity modeling. Elevations of the top of the salt in wells at Keechi Dome range from −85 to −3208 ft MSL.

Keechi Dome is oval in plan view and ridge-like in profile; the long axis trends north-south. On the basis of gravity modeling, investigators have interpreted a slight salt overhang on the south flank at elevation −2000 ft MSL. Estimated horizontal area is 1180 acres at −2000 ft MSL, 1385 acres at −2500 ft MSL, and 2070 acres at −3000 ft MSL.

Oakwood Dome. Oakwood Dome (Fig. 7) is located on the Freestone-Leon County line in Texas; the dome

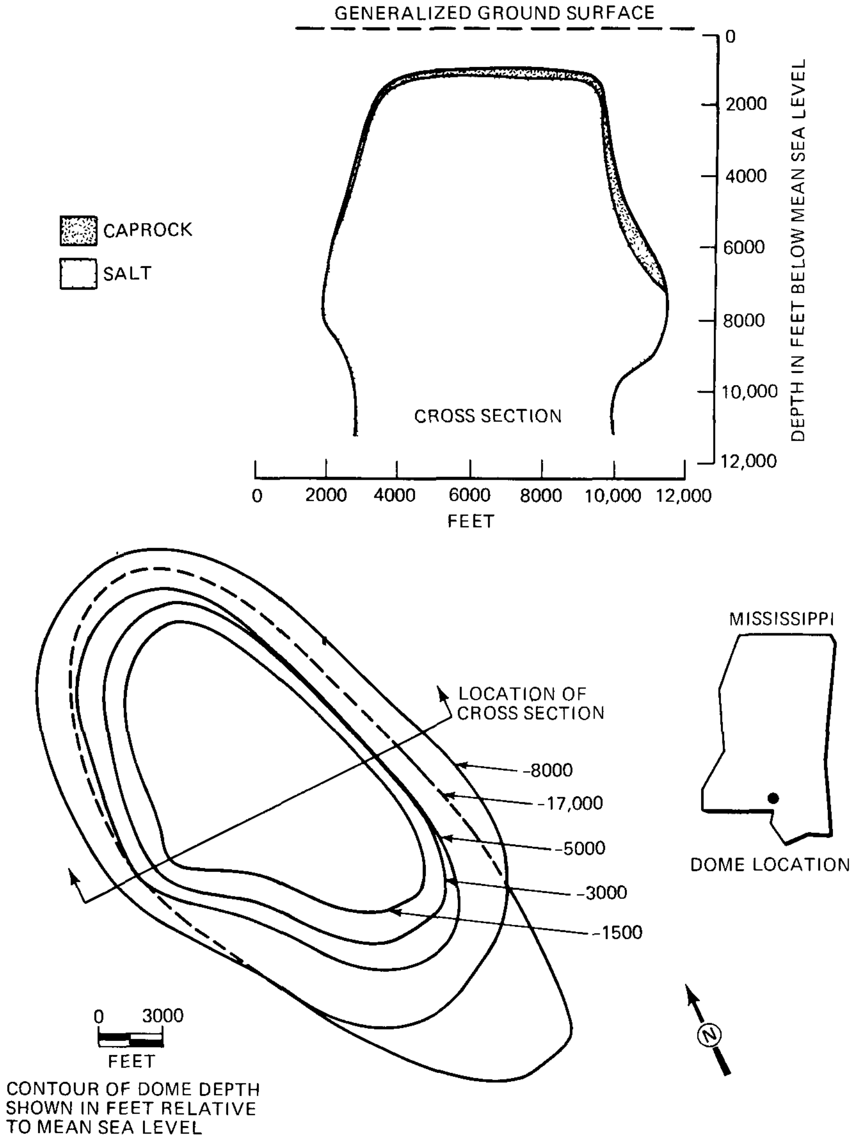

Figure 4 Lampton Dome, Mississippi.

is nine miles northeast of the town of Buffalo and 1.5 miles northwest of the community of Keechi. Most of the land in the dome area is used for farming and livestock grazing; the remaining portion of the land is wooded. The central area of the dome contains an irregular ridge of elongated hills with relief of up to 250 ft. Oakwood Dome is drained by Alligator Creek which flows to the south into the Trinity River.

Elevations of the top of caprock in wells at Oakwood Dome range from −357 to −1695 ft MSL. Caprock varies in thickness from 20 to 450 ft. Well data indicate that the caprock extends down the flanks around the

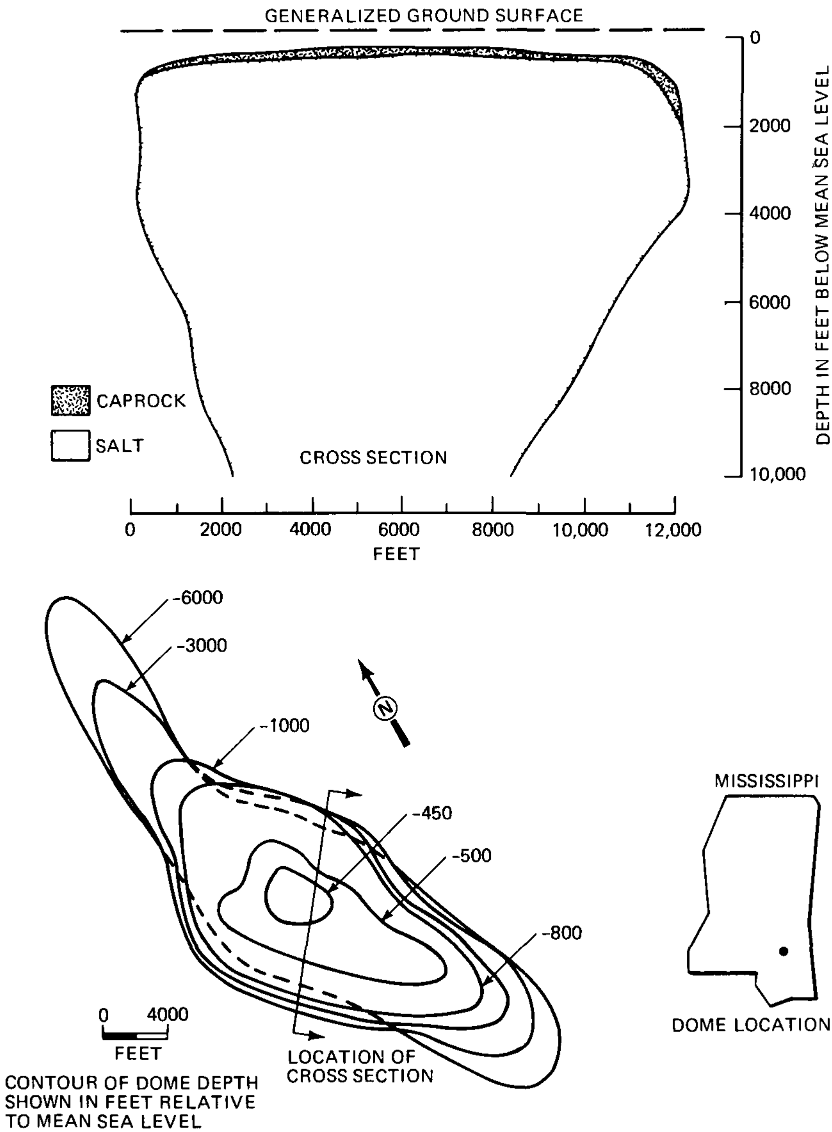

Figure 5 Richton Dome, Mississippi.

salt overhang. Elevations of the top of the salt in wells at Oakwood Dome range from −811 to −1705 ft MSL.

Oakwood Dome is roughly circular in plan view. An overhang (indicated by well data and gravity modeling) that occurs on all sides and ranges from −1000 ft MSL to −5000 ft MSL gives the dome's profile the appearance of a mushroom. Exploration drilling for oil and gas beneath the overhang has resulted in 36 boreholes penetrating the caprock; thirty-five of these boreholes extend into the salt. Thirty-four of the boreholes into salt were sidetracked (whipstocked); as a result, a total of 77 holes were produced in the sediments below the

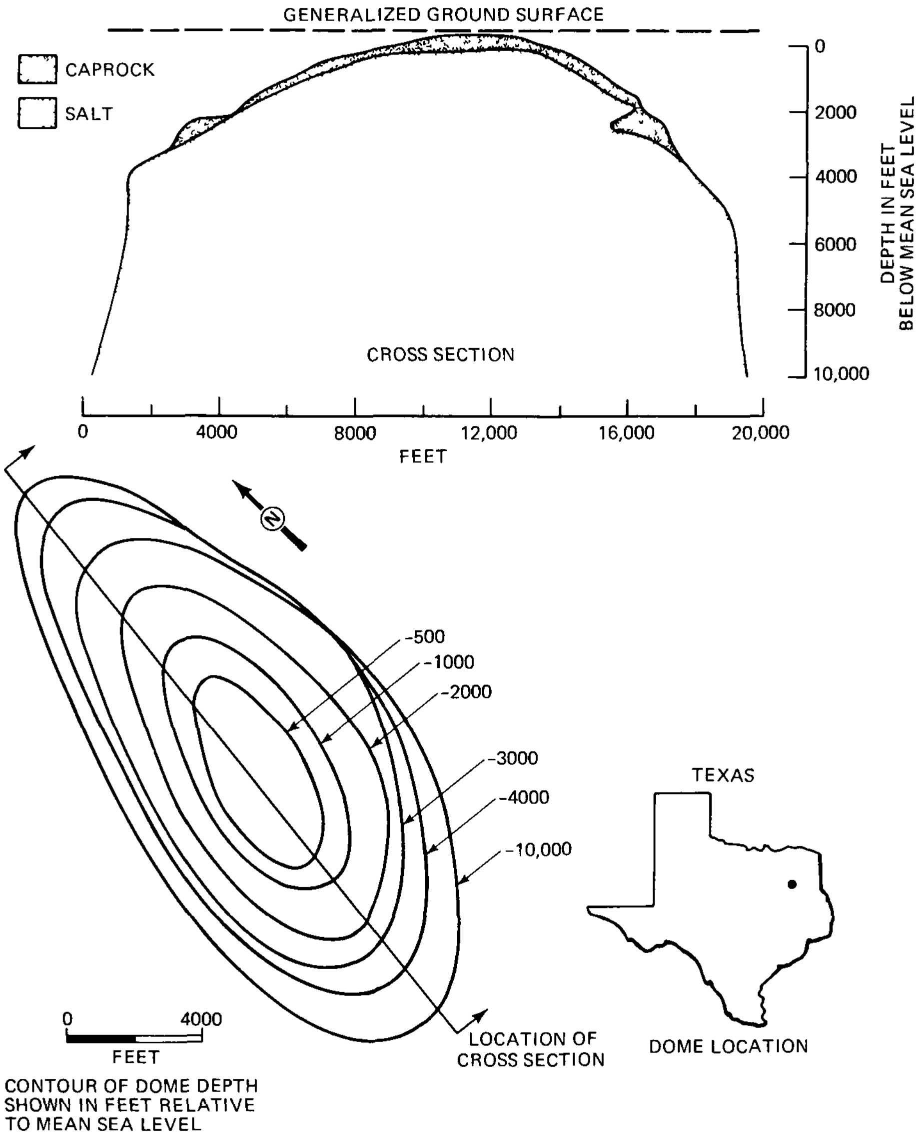

Figure 6 Keechi Dome, Texas.

overhang. These 77 holes reduce the usable area of the dome. Twenty-four of the wells at Oakwood Dome have produced oil or gas. Estimated horizontal area at Oakwood Dome is 2785 acres at −2000 ft MSL, 2694 acres at −2500 ft MSL, and 2613 acres at −3000 ft MSL.

Palestine Dome. Palestine Dome (Fig. 8) is located in western Anderson County, Texas, 4.5 miles west of the city of Palestine. Most of the land in the dome area is thickly wooded; a portion of the land is pasture used for livestock grazing. The dome area is characterized by a central depression

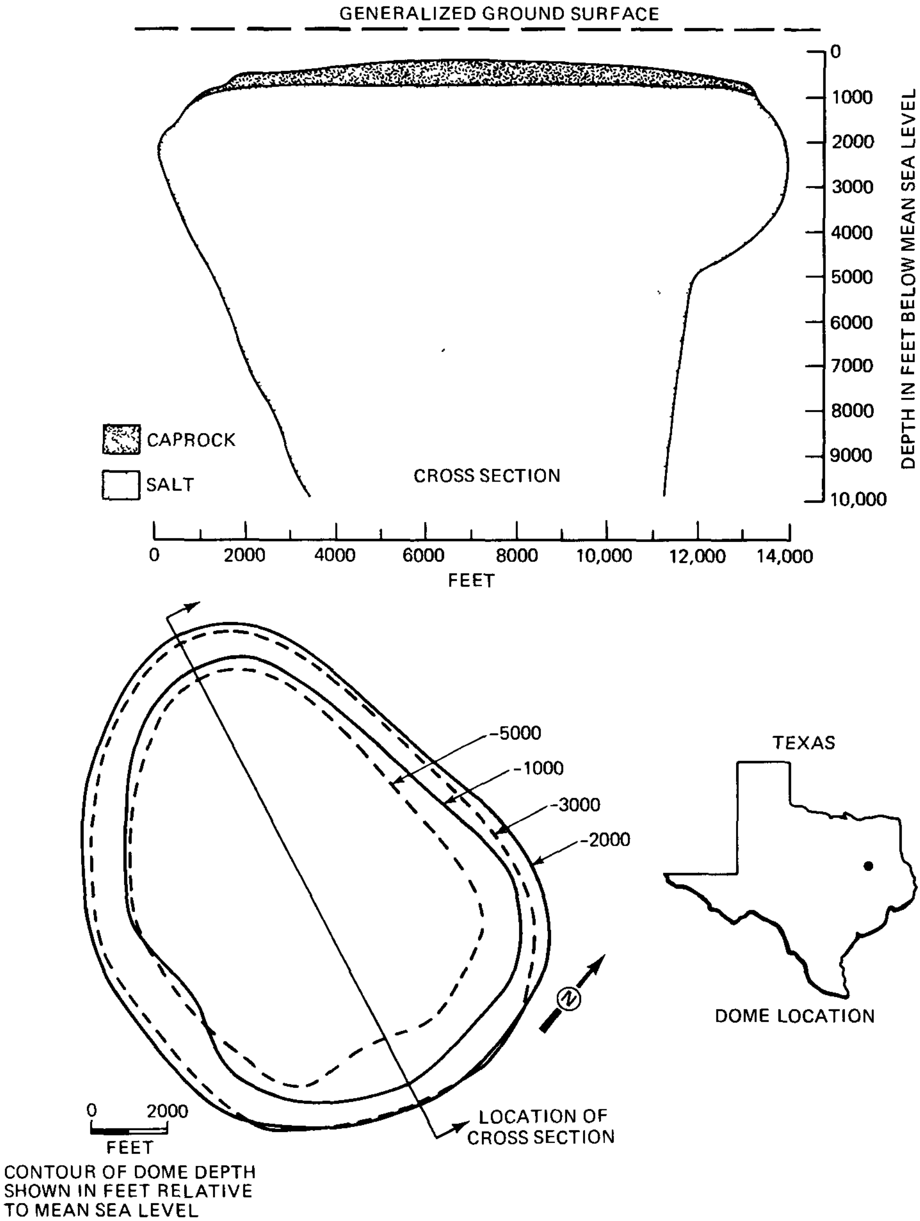

Figure 7 Oakwood Dome, Texas.

surrounded by a series of poorly defined ridges with relief of up to 150 ft. Duggey's Lake occupies the central depression in an area that probably collapsed because of dissolution of the salt. Town Creek and Wolf Creek drain the dome area and flow into the Trinity River.

Elevation of the top of the caprock at Palestine Dome is approximately +200 ft MSL, and the caprock is approximately 50 ft thick. Elevation

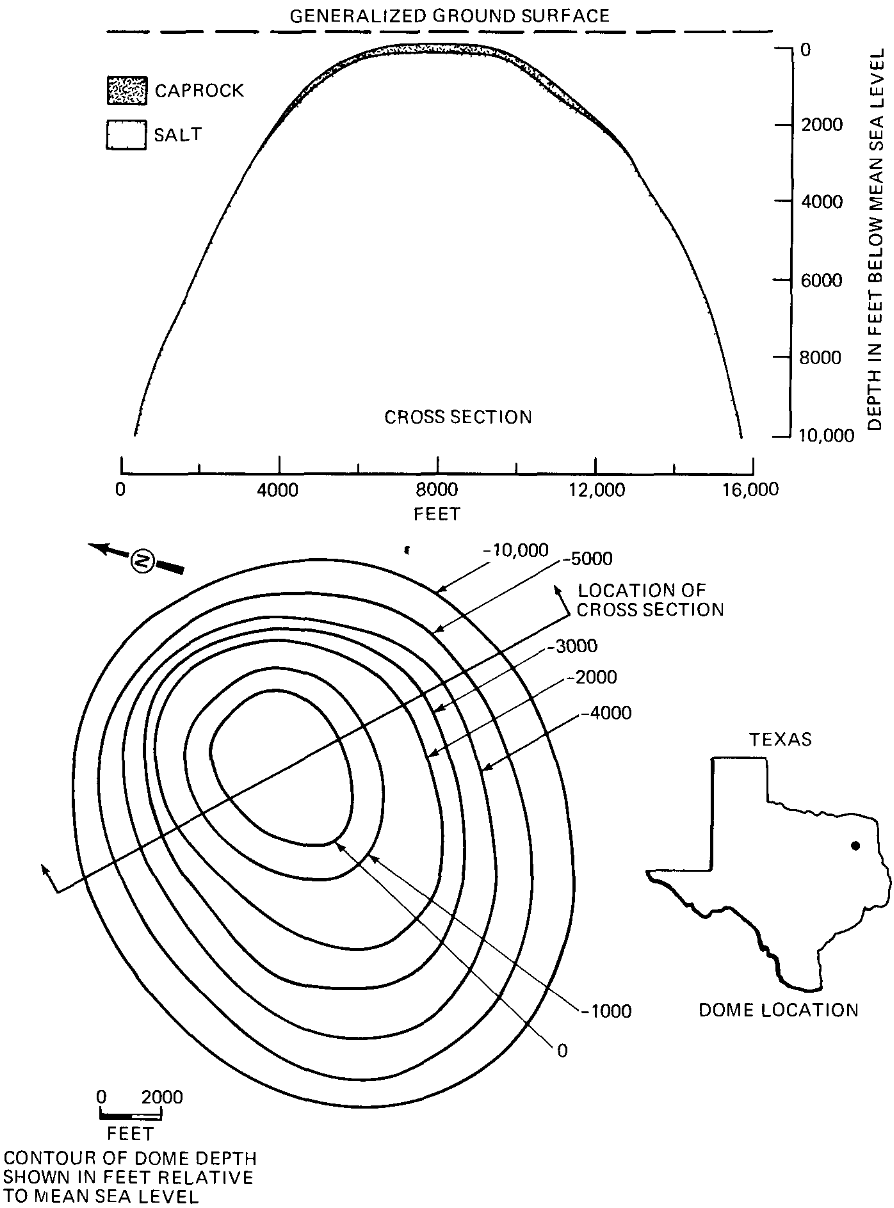

Figure 8 Palestine Dome, Texas.

of the top of the salt at Palestine Dome is approximately +130 ft MSL. The dome is oval in plan view, and well data indicate a possible overhang on the east flank of the dome. How many borehole penetrations are in the dome is not known, but at least 15 brine wells were drilled into caprock and possibly into salt in the early 1900's. Estimated horizontal area of Palestine Dome is 1672 acres at −2000 ft MSL, 1943 acres at −2500 ft MSL, and 2284 acres at −3000 ft MSL.

Regional Characterization

Selecting suitable domes for study was a stepwise region-to-area screening process. This process began with the initial identification of more than 500 salt domes—263 of which were onshore—in the Gulf Coastal Plain. All offshore domes were rejected from further consideration; the interior basins were chosen because they have less potential for hydrocarbon production. In 1977 the Office of Waste Isolation (OWI), Union Carbide Corporation, selected 125 domes in the three interior salt basins for regional study (Office of Nuclear Waste Isolation, 1979). The OWI subcontractor, the Geologic Project Manager (GPM), used established regional geologic screening specifications (Brunton, Laughon, and McClain, 1978) to identify domes that appeared to have high potential as nuclear waste repositories. Seventeen criteria and associated specifications were provided. However, three criteria dominated the screening:

- Domes should be at depths less than 3000 ft.
- Domes should have repository cross-sectional areas greater than 1000 acres in lateral extent, including a surrounding 500-foot salt barrier zone.
- Domes should not have a history of use by industry for hydrocarbon production, storage, or other mineral-related use.

The GPM identified 11 domes that exhibited the highest geologic potential. These domes were Palestine, Brooks, Keechi, Mt. Sylvan, Oakwood, and Boggy Creek in Texas; Cypress Creek, Lampton, and Richton in Mississippi; and Rayburn's and Vacherie in Louisiana.

To optimize the economics and management of acquiring the information necessary to identify suitable repository sites, the Regulatory Project Manager (RPM) evaluated the 11 domes. After considering the likelihood of land-use conflicts and the potential problems in licensing based on nongeologic criteria (NUS Corporation, 1978), the RPM removed three of the 11 domes from further consideration. Domes eliminated were: Brooks, located under Lake Palestine; Mt. Sylvan, near the Tyler urban area and airport; and Boggy Creek, located under the Neches River.

The next objective of the region-to-area screening was to select two or three domes in each basin for additional study.

The combined RPM/GPM evaluation (Office of Nuclear Waste Isolation, 1978) resulted in the recommendation of the following eight salt domes for additional studies during the area characterization phase (Fig. 9):

- Rayburn's Dome, Louisiana
- Vacherie Dome, Louisiana
- Cypress Creek Dome, Mississippi
- Lampton Dome, Mississippi
- Richton Dome, Mississippi
- Keechi Dome, Texas
- Oakwood Dome, Texas
- Palestine Dome, Texas

Area Characterization

After an overall geologic assessment of salt domes in the Gulf Interior Region was completed, investigators selected eight domes for further study in the program's area characterization phase (Law Engineering Testing Company, 1981). Results of this screening evaluation are shown in Fig. 9.

Area characterization activities began in early 1979 in Mississippi, Louisiana, and Texas. The Office of Nuclear Waste Isolation (ONWI); ONWI's GPM, Law Engineering Testing Co. (LETCo); and ONWI's RPM, Bechtel Group Inc.; the Texas Bureau of Economic Geology (TBEG); the Institute of Environmental Studies of

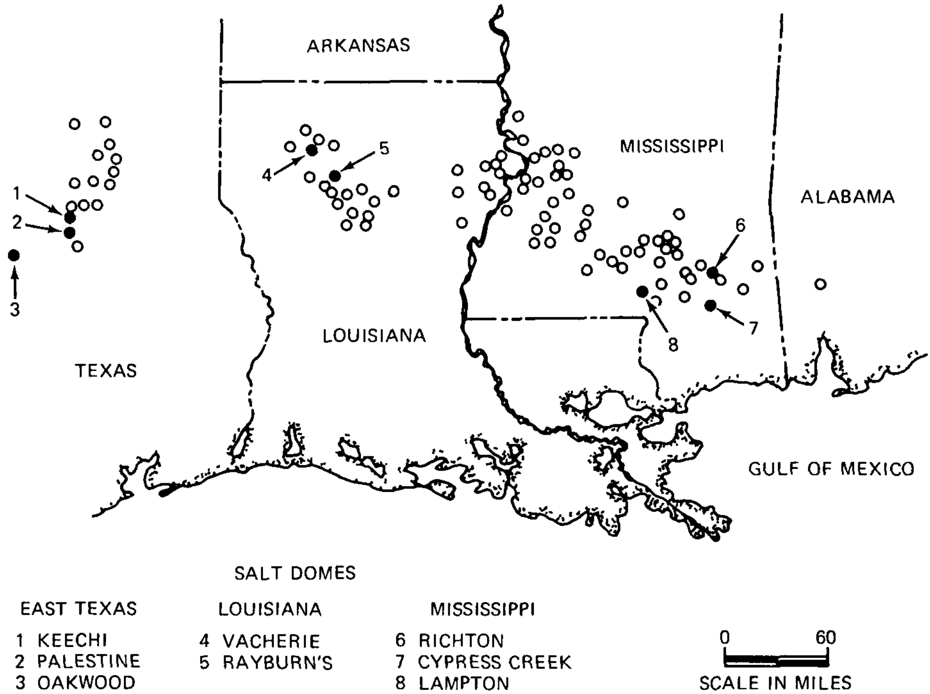

Figure 9 Salt domes under study in Gulf Coast Salt Basins (solid circles) and salt domes rejected in the Basins (open circles).

Louisiana State University (LSU-IES); and the University of Southern Mississippi (USM) conducted field activities, data analysis, and testing activities, and reported upon these activities.

Geologic Characterization. Geologic area characterization involved a number of geological, geophysical, geochemical, and hydrological studies to address National Waste Terminal Storage (NWTS) Program criteria (U. S. Department of Energy, 1981). Each of the studies addressed issues and provided input to resolve questions related to specific issues.

Drilling activities included shallow borings and deep boreholes. Deep drilling (maximum 3500 ft) was conducted to gather data on lithology, stratigraphy, and geochemical and hydrologic properties of formations down to repository depth.

Several boreholes were drilled to obtain cores that were used for laboratory tests. These were samples from caprock, salt, and also sidewall cores of the geologic formations penetrated. Thirty-nine deep holes were drilled in the formations in Mississippi; eighteen deep holes were drilled in the formations in Louisiana; and seven deep holes were drilled in the formations in Texas.

The shallow borings were done to gather data about the near-surface formations (0 to 300 ft) and to help develop the surface geologic maps. Eighty-five shallow borings were made in the formations in Mississippi; 102 shallow borings were made in the formations in Louisiana; and 42 shallow borings were made in the formations in Texas.

Downhole Geophysical Logs. A suite of logs was run in the boreholes at each site to define stratigraphy, lithology, and other soil and rock properties. This suite of logs is intended to provide data on representative lithology (grain size and sand vs. clay relations), stratigraphy (stratigraphy and thickness of units correlated against core log), structure (stratigraphic relations pertaining to potential faulting, folding, and hydrologic "short circuits" through the sediment layers), and hydrology (permeability, porosity, and salinity). The types of logs include the following:

- Dual induction focused
- Single point resistance
- Spontaneous potential
- Short normal
- Long normal
- Caliper
- Gamma ray
- Acoustic
- Compensated neutron
- Temperature
- Dual lateral
- Acoustical televiewer
- Density

Gravity Studies. Gravity surveys were used to define the sizes and shapes of the domes. The gravity field at the surface, which varies as a result of density differences among the salt, the caprock, and the surrounding sedimentary rock, was measured as discrete points. These data, combined with borehole data, were used to generate a three-dimensional model of each dome. Accuracy of this model is controlled by the number and distribution of the gravity data points and the amount of borehole control. In the area phase, data were sufficient to approximately define the depth and lateral extent for each dome.

Seismic Reflection. Another geophysical technique that was used extensively in the Gulf Coast area is seismic reflection. Seismic waves produced by explosive or mechanical devices were propagated through the subsurface strata, and the reflections were recorded by transducers on the surface. When the data were processed, they yielded a representation of the subsurface that was used to locate and define the domes and surrounding strata.

Substantial Common Depth Point (CDP) seismic reflection data were purchased during the area phase. These surveys, already performed by companies seeking oil and gas prospects, were easily obtained. High-resolution seismic reflection (HRSR) surveys were made over the domes to define the caprock and the edges of the domes; these surveys were subcontracted and were subject to limitations of land access. Nevertheless, in several cases HRSR surveys provided additional input for the gravity models; in fact, the interpretations of the sizes and shapes of two of the domes were substantially changed as a result of data provided by the HRSR surveys.

Hydrologic Testing. Tests performed on hydrologic holes included aquifer pumping tests, flow tests on artesian wells, and drill stem tests. Laboratory permeability tests were performed on sidewall cores. Transmissivity, hydraulic conductivity, the storage coefficients, and the density of water were determined for each location. The hydraulic head measurements made in the boreholes were corrected for density to depict the true potentiometric surface, and the potentiometric surface maps were used to determine flow paths.

Water samples were taken to study the geochemical properties, age, and isotopic composition of the water. This information also was used to better define the flow paths of the groundwater.

Hydrologic boreholes are now being regularly monitored to obtain a data base for determining long-term water level trends and for evaluating changes resulting from stresses imposed on the system by man and nature.

Seismicity. The earthquake potential of the Gulf Coast salt dome region generally is regarded as extremely low because of the lack of historical seismicity and the absence of crustal plate boundaries. Therefore, in the area characterization phase of investigations, the seismicity studies were mainly literature searches to document past activity.

Because of the scarcity of historical data and the lack of an established seismographic network, researchers used portable microearthquake recorders. These systems, which consist of a low-frequency geophone and a paper drum recorder, can be moved easily to survey large areas or to monitor activity.

Because of past activity in the East Texas Basin, researchers positioned a single microearthquake station near the Mt. Enterprise Fault System to record the activity. Sufficient activity was recorded in the initial period of operation to warrant a larger network of stations. Microseismic data are needed in several additional areas to address past seismic activity. These studies will be used to determine the locations and causes of the events and the maximum credible event.

Geologic Mapping. Surface and subsurface geology have been mapped to define the lithologies, stratigraphy, and structure of the area. The maps were used to delineate aquifer recharge areas and to characterize the structure in the dome areas.

Geologic Area Characterization Reporting. Geologic area characterization reports contain a summary of all data gathered during the area characterization phase (1979 to 1981) (Law Engineering Testing Company, 1982, 1982a, 1982b, 1982c).

Volume I (Law Engineering Testing Company, 1982) contains a history of the program, describes the participants in the various phases of the program, and presents a short summary of the area characterization program. The technical reports (Law Engineering Testing Company, 1982a, 1982b, 1982c) are broken down by geographic area and include the following sections:

- Geologic Characterization of Study Area
- Geologic Characterization of Selected Domes
- Hydrologic Characterization of Study Area

The texts present the data collected during the area phase; they contain tables and maps constructed from the data, maps of locations of field activities, cross sections of the study area, hydrologic figures, geologic figures, and other information in graphical form. The raw data and calculations used to derive the information are contained in appendices.

Additionally, topical reports that discuss subjects of particular interest will be published. Topics to be covered include potential erosion and inundation, mineral resources, geochemistry of salt, geothermal characteristics of domes, permeability of sedimentary rocks surrounding domes, and tectonic features.

Environmental and Socioeconomic Characterization. Environmental and socioeconomic characterizations for the area phase were based on literature surveys of data available from state and federal agencies. The characterizations describe the surface water, atmosphere, background radiation, demography, socioeconomic parameters, land use characteristics, and ecosystems (including agricultural systems) in each of the three study areas. Specific topics include:

- Runoff
- Floods
- Water quality

- Water use
- Meteorology
- Air quality
- Background radiation
- Population density and distribution
- Employment
- Income
- Housing
- Community services
- Land use and ownership
- Cultural resources
- Transportation
- Parks and monuments
- Terrestrial ecology
- Wildlife refuges
- Threatened and endangered species
- Agricultural systems
- Aquatic ecology

Surface Water. Surface water characteristics are important to assessing site drainage, to assessing the potential for surface water contamination, and to evaluating the potential for flooding of surface facilities. Several small lakes and intermittent streams are located in and around the domes. Streambeds consist predominantly of sand, silt, and gravel. Average runoff into the river basins in which the domes are located ranges from 11 to 26 inches/mi^2 annually. A majority of runoff occurs between January and April when local flooding is the heaviest. The water generally is soft and low in mineral content. There is some industrial use of surface waters and some pollution of surface waters from municipal and industrial sources.

Atmosphere. Atmospheric conditions, including severe weather, may affect repository operations or cause environmental impacts related to repository construction and operation; however, within the areas evaluated, there is little variation in meteorological conditions. The climate, which is controlled primarily by subtropical latitude and proximity to the Gulf of Mexico, generally is humid with a short cold season and a long warm season. Mean annual temperatures range from 65 to 67°F; annual precipitation ranges from 48 to 61 in. Average wind speeds are low; they range from 5.5 to 9 mi/h. The area experiences tornadoes and severe thunderstorms during all months of the year; hurricanes occur, but they generally are weakened in intensity as they move inland from the Gulf of Mexico.

The domes are located in predominantly rural areas and are removed from sources of industrial pollution. The flat terrain with low rolling hills permits unobstructed air flow and atmosphere dispersion that contribute to the generally favorable air quality. All domes are located in areas that the Environmental Protection Agency considers attainment areas.* The closest Class I† area to any of the domes is 92 mi away.

Background Radiation. Background radiation data provide perspective on the potential radiation doses to individuals from sources associated with a repository. Because of their low elevations, areas where the domes are located receive low levels of cosmic radiation. No specific data for the areas under consideration are available to closely characterize natural terrestrial radiation; specific fallout data for the areas also are lacking.

Socioeconomic Investigations. Demographic, socioeconomic, and land-use patterns indicate the quality,

**An attainment area is an area judged by the Environmental Protection Agency (EPA) to be equal to or better than the primary or secondary National Ambient Air Quality Standards (NAAQS). Primary standards establish levels of air quality that the EPA judges necessary to protect the public health; secondary standards establish levels of air quality judged necessary to protect the public welfare from any known or anticipated adverse effects of a pollutant (42 USC 7473 and 7501, Clean Air Act, as amended).*

†Class I means that increases in baseline air quality with respect to ambient sulfur dioxide or particulate matter must be restricted to very low percentages of the corresponding NAAQS.

type, and extent of human activity in the study area. This information provides a basis for assessing the socioeconomic impacts that a repository may have on people and their activities. Population densities indicate the relative potential impact on humans of radiation from the transportation and storage of nuclear wastes. Population also can be used as an indicator of the level and type of services available in the area. Socioeconomic parameters, including employment participation rates, labor force characteristics, and income, are indicators of the nature of an area's economy; these parameters can be used to analyze a community's potential ability to provide required labor for the development and operation of a repository. Income indirectly measures the strength of an area's economic base; other economic indicators include housing stock, health care facilities, and planning functions. An evaluation of existing and past land use is important in determining a compatible location for a nuclear waste repository. Knowledge of existing transportation networks can indicate the ability of the area to meet needs associated with the construction and operation of a repository. The domes under consideration are located in predominantly rural areas with low population densities. Employment participation rates* within counties where the domes are located range from 29 to 33 percent. Major employment is in manufacturing and in wholesale and retail trade; local incomes are considerably below the national average. Land on the domes is primarily privately owned and in forestry and agricultural uses. All domes are within five miles of a U. S. or state highway and within ten miles of a rail transportation system.

Ecology. Information on ecosystems, both natural and agricultural, is important for identifying threatened or endangered species, for assessing potential impacts, and for evaluating future changes that may result from the construction and operation of a repository. The domes are located within the loblolly shortleaf pine, longleaf-slash pine, and post oak savannah/piney woods plant communities. The white-tailed deer is the most common large mammal. Although rare, the black bear occasionally is encountered in the longleaf-slash pine and post oak savannah/piney woods communities. Other animals associated with these plant communities include the cottontail, raccoon, opossum, squirrel, and fox. Turkey, bobwhite, mourning dove, and quail are the predominate terrestrial game birds. Potential habitats of threatened or endangered species are at all domes; however, the existence of such species at any of the domes has not been verified.

The domes under consideration are covered with a mix of forested and agricultural land. Trees harvested commercially are primarily loblolly, longleaf, and shortleaf pine; harvested hardwoods include oak, gum, hickory, maple, and elm. Hay, corn, and soybeans are the major cultivated crops; some beef production occurs in the dome areas, and a few small dairies also are present. On and around the domes are various small intermittent and perennial creeks that typically are surrounded by bottomland hardwood forest. Fish frequently found in the streams and creeks include catfish, sunfish, and crappie. No endangered or threatened aquatic species are listed for the dome areas.

Environmental and Socioeconomic Characterization Reporting. The following environ-

**The total number of persons in the labor force in an area as a percentage of the total area population. The labor force includes both the employed and those seeking employment.*

mental area characterization reports have been completed by Bechtel Group, Inc. for each study area:

- *Environmental Characterization Report for the Gulf Interior Region, Louisiana Study Area* (Bechtel Group, Inc., 1982)
- *Environmental Characterization Report for the Gulf Interior Region, Mississippi Study Area* (Bechtel Group, Inc., 1982a)
- *Environmental Characterization Report for the Gulf Interior Region, Texas Study Area* (Bechtel Group, Inc., 1982b)

Area Screening for Location Phase

With completion of the area characterization phase and the accompanying reports, ONWI completed a process of screening that reduced the number of salt domes subject to further study from eight to four. These four domes will be the object of further screening efforts in the location phase that will begin in fiscal year 1983. The report *Evaluation of Area Studies of the U. S. Gulf Coast Salt Dome Basins* was submitted to the public for review and comment. Comments were received from the states. The comments were addressed, and a final document was released in 1982 (Office of Nuclear Waste Isolation, 1982). This evaluation documents the decision process, supportive data, and actual selection of the four domes. The four domes selected were:

- Richton Dome in Perry County, Mississippi
- Cypress Creek Dome in Perry County, Mississippi
- Vacherie Dome in Bienville and Webster Parishes, Louisiana
- Oakwood Dome in Leon and Freestone Counties, Texas

Domes were eliminated when there was a high likelihood that major siting criteria would not be met. Screening decisions were made in order to focus efforts on the most favorable and acceptable locations.

Approach. The general approach used for screening consists of the following steps:

- Identify factors (criteria and subcriteria) and supporting information needs important to the screening decision.
- Gather the required information in accordance with applicable consultation procedures.
- Identify candidate locations (domes) from area studies.
- Evaluate each candidate location (dome) according to the previously identified factors.
- Compare and recommend candidate locations (domes).
- Review the screening decisions in accordance with applicable consultation procedures.

Criteria used for this screening activity were produced by DOE and published as *NWTS Program Criteria for Mined Geologic Disposal of Nuclear Waste: Site Performance Criteria,* DOE/NWTS-33(2) (U. S. Department of Energy, 1981). The criteria are listed below:

I. Site Geometry
   a. Thickness of host rock
   b. Lateral extent of host rock
   c. Depth to host rock
II. Geohydrology
   a. Geohydrologic regime/flow
   b. Hydrologic regime/modeling
   c. Geohydrologic regime/shaft construction
   d. Dissolution
III. Geochemistry
   a. Chemical interaction
   b. Radionuclide retardation
IV. Geologic Characteristics
   a. Stratigraphy
   b. Host rock
V. Tectonic Environment
   a. Tectonic elements
   b. Quaternary faults

c. Quaternary igneous activity
d. Uplift/subsidence
e. Seismicity

VI. Human Intrusion
a. Resources
b. Exploration history
c. Land ownership/control

VII. Surface Characteristics
a. Surficial hydrologic system
b. Surface topography
c. Meteorological phenomena
d. Industrial, transportation, and military installations

VIII. Demography
a. Population density/urban areas
b. Radioactive waste transportation risk

IX. Environmental Protection
a. Environmental impact
b. Air, water, and land-use conflicts
c. Normal and extreme environmental conditions

X. Social/Economic Impact
a. Social/economic impacts
b. Transportation, access, and utilities

Dome Suitability Evaluations. The eight domes studied in the area characterization phase were rated with respect to each criterion and subcriterion of *NWTS Program Criteria for Mined Geologic Disposal of Nuclear Waste: Site Performance Criteria*, DOE/NWTS-33(2) (U. S. Department of Energy, 1981). The domes were rated most favorable, acceptable, less favorable, or eliminated. As a result of this screening, ONWI recommended the following four domes for location phase screening: Richton, Vacherie, Cypress Creek, and Oakwood.

Richton Dome was rated acceptable and most favorable because its larger lateral extent would provide a very large buffer zone and technically conservative repository loadings. However, Richton is potentially less favorable because of a land-use and socioeconomic conflict, i.e., the location of the town of Richton within a potential control zone for the repository site.

Vacherie Dome was rated acceptable; however, it has no significant advantages and is somewhat less favorable because of the surface hydrology and apparent dissolution of the host rock.

Cypress Creek Dome was rated acceptable. However, although the existing land-use factor was rated more favorable than, the overall rating of the dome was less favorable than because of its resources, geochemical regime, potential dissolution, and surface hydrology.

Oakwood Dome, although rated acceptable for the purpose of this evaluation, has been assessed as much less favorable because of its exploration history and significant petroleum exploration. Approximately 36 holes have penetrated the salt. Some of these holes have penetrated to repository level, and many have been further whipstocked beneath the salt overhang. In addition, the proximity of potential Quaternary faulting, dissolution, and the surface hydrology conditions are assessed less favorable.

Palestine Dome in Texas was eliminated in 1979 because it had a significant safety flaw related to the licensing process and associated with prior dissolution processes.

Keechi Dome was eliminated because of its inadequate lateral extent and inadequate depth.

Lampton Dome was eliminated because its lateral extent is inadequate.

Rayburn's Dome was eliminated because of inadequate lateral extent with regard to reference repository loading and inadequate minimum depth.

Site Characterization Planning for Location Phase

Planning for location-phase screening studies is underway on the four domes selected for location studies. This effort is being directed by ONWI. Department of Energy contractors and designated state representatives are working with ONWI and subcontractors to produce a characterization plan for the four domes. Past meetings with the states have elicited a number of issues of concern in the three states in conjunction with issues identified in *Evaluation of Area Studies of the U. S. Gulf Coast Salt Dome Basins: Location Recommendation Report* (Office of Nuclear Waste Isolation, 1982). These issues and those related to *NWTS Program Criteria for Mined Geologic Disposal of Nuclear Waste: Site Performance Criteria* (U. S. Department of Energy, 1981) will provide the basis for screening studies in the location phase. Following location studies, a salt dome will be compared with bedded salt sites for selection of a first exploratory shaft site.

Provided in the following sections is a partial listing of issues that have been identified by the states, presented to DOE and ONWI, and related to *NWTS Program Criteria for Mined Geologic Disposal of Nuclear Waste: Site Performance Criteria.*

Vacherie Dome, Louisiana.

Technical Issues by Criteria.

1. Site geometry—lateral dimensions of the dome, including buffer zone.
2. Geohydrology—potential dissolution of the salt stock and the near-dome hydrologic regime.
3. Tectonics—structure of over-dome sedimentary units.
4. Environment—potential for an endangered species and a wild and scenic river near the dome.

Legal and Institutional Issues by Criteria.

1. Socioeconomic-human resources (labor force availability) and the economic base and infrastructure of communities near the dome.
2. Land use-land availability and the potential for its acquisition for the repository.

Cypress Creek and Richton Domes, Mississippi

Technical Issues by Criteria.

1. Geohydrology—potential dissolution of the salt stock; this potential may be identified by salinity anomalies through determination of hydrologic flow.
2. Tectonics-potential for seismicity resulting from the Phillips Fault Zone to the north-northeast.
3. Environment—potential for endangered species over and near the domes.

Legal and Institutional Issues by Criteria.

1. Socioeconomic—economic and social impact upon local communities.
2. Land use—recreation and groundwater use by local communities.

Oakwood Dome, Texas.

Technical Issues by Criteria.

1. Geohydrology—the identification of the near-dome hydrology regime, existing oil wells through the dome overhang as related to its exploration history for petroleum.
2. Tectonics-reported near dome Quaternary faults and structure of overdome sedimentary layers.

Legal and Institutional Issues by Criteria.

1. Socioeconomic—economic and social impact upon local communities.

2. Land use—potential air, water, and land-use conflicts and an inventory of groundwater usage.

When further work is undertaken on the remaining four domes, investigators will initiate limited field work related to socioeconomic impact and environmental protection. Information resulting from this work will be used to aid in selection of a site for an exploratory shaft and to build the environmental data base required for the environmental document [Environmental Assessment (EA) or Environmental Impact Statement (EIS) required for shaft construction].

The socioeconomic impact assessment will include:

- Evaluation of existing and projected fiscal capacity of host jurisdiction, including revenue sources, expenditures, and interrelationships of taxing jurisdictions.
- Evaluation of existing infrastructure, including public and private services and facilities (police, fire protection, sewer and water services, housing, education, utilities).
- Evaluation of present and projected land uses, land-use regulations and policies, and anticipated changes in land use and/or land values.

The environmental protection work will consist of:

- Limited field surveys to identify the presence of any threatened or endangered terrestrial or aquatic species and/or critical habitats.
- Surface archaeological surveys to determine the presence or absence and significance of archaeological and historical resources, including prehistoric sites.

Information also will be obtained to document the acceptability of the site relative to applicable federal, state, and local regulations and laws.

Eventually, a complete 52-week environmental baseline program necessary for a license application will be initiated during the detailed site characterization phase. This program will include on-site meteorological investigations, field surveys of plants and animals, water quality surveys, background radiation surveys, and noise surveys.

Future Activities

Activities anticipated for the future are preparation of the site characterization report for addressing location phase issues, distribution of the final area characterization reports and the location recommendation report, and revision of the site characterization plans to address comments submitted by state and federal reviewers.

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Hydrologic Issues in Repository Siting

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Extrapolation of Darcy's law to the transport of water and solutes in unfractured poorly permeable rocks being studied for nuclear waste disposal is questioned. The hydrologic literature includes numerous references to both non-Darcian flow in dense materials devoid of macrofractures and microfractures and to threshold gradients below which no flow occurs. For such situations to occur, the pore-size range must be small enough so that all pore water is sufficiently close to mineral surfaces to be affected by the surficial forces. Then the flow will be non-Newtonian and non-Darcian, and solute transport will be by molecular diffusion.

If fluid transport in very dense unfractured rocks is non-Darcian, useful methods of testing candidate host rocks become apparent. In situ nondestructive pressure testing of canister waste emplacement boreholes in a mined repository can verify the absence of both fracture flow and Darcian flow. 18 ref.

Introduction

The most likely natural process by which radionuclides might reach the biosphere from a deep nuclear waste repository is subsurface hydrologic transport. Consequently substantial effort is being devoted to problems of subsurface hydrologic transport. Significant advances are being made in the understanding of regional hydrogeology and of the hydrologic characteristics of poorly permeable rocks. However, it is possible that misapplication of Darcy's law to the transport of water and solutes in unfractured poorly permeable rocks is giving an erroneous perception of the rocks' containment capabilities.

One group of questions central to this concern relates to the applicability of laboratory test results and Darcy's law to water transport in dense poorly permeable clays and rocks that are essentially devoid of macrofractures and microfractures. Additional questions relate to the mechanism of transport of radionuclides in whatever non-Darcian flow fields that may be present in these materials. These uncertainties are in addition to the uncertainties relating to geochemical questions that involve sorption of radionuclides on surfaces.

Answers to the hydrogeological questions probably will indicate that current requirements for exploration, siting and design of nuclear waste repositories are very conservative. Furthermore, although excellent planning, exploration, and research are in progress on a number of waste-disposal projects, the overall strategies and requirements for hydrologic field testing and design are not necessarily developing along technically optimal directions. Finally, preoccupation with advanced technology may be obscuring consideration of some simple but effective approaches such as nondestructive pressure testing of waste containment boreholes in the mined repositories.

The uncertainties outlined above are not unexpected. Groundwater hydrology has been developed for use with granular permeable rocks capable of providing water supplies. The concern in nuclear waste disposal is with the capabilities of relatively impermeable rocks to contain and immobilize wastes. In fact if some of the available technology, such as pumping tests, can be applied to a potential host rock, that rock is too permeable for use in repository siting.

Another reason for the uncertainties discussed above is that the siting and design of nuclear waste repositories have involved time frames of lengths that can permit substantial climatic and geologic evolution. Fortunately for the designer, shorter time frames are now being considered. For example, the Nuclear Regulatory Commission (1980) indicates that, "The Department (of Energy) shall emphasize the first 10,000 years following decommissioning in their prediction of changes in natural conditions and the performance of the geologic repository." Hydrologic design and prediction are feasible at geologically simple and stable sites over such time frames, which are relatively short from the geologic point of view.

We are concerned that standard concepts and methods dealing with groundwater have been applied to these unfamiliar hydrologic systems without sufficient questioning. Therefore the purpose of this paper is to ask questions rather than to provide answers. We believe that the consequences that are possible in the event that our misgivings are well-founded justify raising these concerns even if they prove to be unfounded.

Prediction of Radionuclide Transport

Use of the models and methods available for predicting solute transport from a repository involves the solution of partial differential equations describing the transport of water, solutes, and heat. The first equation is for confined subsurface saturated flow. By using the notation of Bear (1972), investigators can write the following equation describing single-phase flow in a saturated porous medium in three dimensions:

$$\nabla \cdot \frac{\rho \vec{\kappa}}{\mu} (\nabla P - \rho g \nabla z) - q = \frac{\partial}{\partial t}(\phi \rho) \quad (1)$$

where ρ = density of the fluid
q = mass rate of fluid injection or withdrawal
t = time
ϕ = porosity
$\vec{\kappa}$ = intrinsic permeability
z = elevation above a reference plane
P = pressure
μ = dynamic viscosity
g = acceleration due to gravity

Given the initial and boundary conditions; the values of q, ϕ, ρ, and μ; and the components of the intrinsic permeability tensor, $\vec{\kappa}$, Eq. 1 can be solved analytically or numerically for the hydraulic head distribution in one, two, or three dimensions as a function of time. The values of hydraulic head are then used with Darcy's law to determine the groundwater velocity components.

A form of the equation for solute transport in a porous medium subject to radioactive decay and linear sorption is (Bear, 1972):

$$\frac{\partial C}{\partial t} = \frac{\partial}{\partial x_i}\left(\frac{D_{ij}}{K_f} \frac{\partial C}{\partial x_j}\right) - \frac{u_i}{K_f} \frac{\partial C}{\partial x_i} - \lambda C \quad (2)$$

where C = concentration
t = time
D_{ij} = coefficient of dispersion
u_i = average pore velocity
x_i, x_j = cartesian coordinates
K_f = retardation factor
λ = radioactive decay rate

Analytical or numerical solution of Eq. 2 yields the solute concentration, C, as a function of space and time.

Although an equation for heat transport may be important, it is not essential to the questions raised in this paper, and it is not discussed further.

Even though common practice is to apply Eqs. 1 and 2 to the prediction of radionuclide transport from a proposed repository, several sources of uncertainty are identified with their use.

Equations 1 and 2 are applicable to transport by Darcian flow through porous media. Darcian flow occurs when the macroscopic discharge rate is proportional to the hydraulic gradient, and it requires laminar flow by a Newtonian fluid. In the case of a nuclear repository site underlain by very dense, unfractured, sufficiently impermeable rocks, it is possible that the fluid-flow regime will be non-Darcian. In fact, if Darcian flow can occur in such a material, that material may be too permeable for use as a repository host rock.

In the case of a nuclear repository site underlain by low-permeability sparsely fractured materials such as shales, crystalline rocks, or salt, the geometry of the fractures will determine the nature of the fluid flow and the dispersion of the solutes. Unless the fractures fall within certain width ranges and configurations, the flow will be non-Darcian.

Despite the departures from the conditions required for Eqs. 1 and 2 to be applicable, the equations sometimes are used to analyze transport from prospective repositories. This is done because no alternative, completely developed, usable theory is available for computing groundwater and solute transport in either essentially impermeable rocks or in fractured rocks. Furthermore, questions exist about the validity of some of the parameters obtained from tests of such systems. The net result of these applications of Eqs. 1 and 2 may have been to give erroneous perceptions of the feasibility and the problems of radionuclide containment.

Darcy's Law and Hydraulic Conductivity

Hydraulic conductivity values that are supposed to be smaller than 10^{-10} cm/s are reported from studies of potential repository host rocks. Isherwood (1981), Freeze and Cherry (1979), and Stuart, Brown, and Rhodehamel (1954) are among those reporting numerous examples of such low values. Some of the values are obtained from laboratory determinations made at large pressure gradients that are orders of magnitude greater than natural field gradients. If very large time frames are used, solution of Eqs. 1 and 2 may predict objectionable water and radionuclide transport to the biosphere for even these low values of hydraulic conductivity. Therefore a logical conclusion has been that every repository must leak over the design time frame. Furthermore, it appears that the repository rocks cannot be depended on to exclude groundwater, to prevent the water from reacting with the stored wastes, and subsequently to prevent the transportation of radionuclides to the biosphere. Instead, it appears that the geochemistry must be relied on to immobilize the radionuclides by sorption or other means; however, because of geochemical uncertainties, such sorption may be difficult to predict. Therefore the task of proving containment becomes exceedingly difficult. Results of this sequence of ideas are apparent in 10CFR Part 60 (Nuclear Regulatory Commission, 1980) where some siting requirements may have been stipulated unnecessarily. Thus such use of the transport equations may have led to serious underestimation of the containment capabilities of dense unfractured host rocks.

Darcy's law is a linear relationship between discharge and hydraulic gradient with hydraulic conductivity as the constant of proportionality. It is an empirical relation that has been established for laminar Newtonian flow through granular porous media. There is evidence that Darcy's law does not hold for some fine-grained porous media in the clay range. As expressed by Olsen (1966), "...the evidence as a whole suggests that Darcy's law is obeyed in many natural sediments, but that exceptions may occur in very fine-grained clays, specifically, montmorillonite, and also in shallow unconfined clays or in granular soils containing small amounts of clay."

As discussed by Adamson (1967), "...where interfaces are concerned, there is a large variety of persuasive, but not absolutely conclusive, evidence that disturbances may extend hundreds or thousands of Angstroms from a surface." They are due to electrostatic and Van der Waals forces and "...to perturbations passed on from molecule to molecule." Regardless of the causes, the structure and viscosity of very thin films are anomalous close to surfaces. The flow is non-Newtonian and non-Darcian.

The surface effects may prevail in clay and non-clay materials. As noted by von Engelhardt and Tunn (1955), "It is presumed that the bonding of water occurs chiefly on the surfaces of the clay mineral particles, although it is possible that the quartz surfaces themselves fix a certain portion of the fluid."

The hydrologic literature contains numerous references to non-Darcian flow in fine materials and even to threshold gradients below which no flow occurs. As expressed by Bouwer (1978), "Another situation where Darcy's law may not be valid is where water flows through dense clays. The pores in such materials can be so small that the water molecules in the pores are influenced by the double-layer effects of the clay particles. Because water molecules are polar, water near the electrically charged clay particles then has a more crystalline or 'icelike' structure, which causes the viscosity to be higher than that of 'free' water. Under such conditions, small hydraulic gradients may not be sufficient to produce water movement, giving rise to threshold gradients and a nonlinearity between flow rate and hydraulic gradient, in contrast with Darcy's law (Swartzendruber, 1969). Other phenomena that cause flow through dense clays to deviate from Darcian flow include movement and rearrangement of clay particles due to frictional drag by the flowing water, development of electrokinetic streaming potentials, and electroosmotic counterflow (Kutilek, 1972; Elnaggar, Karadi, and Krizek, 1974)."

Bear (1972) also discusses "a lower limit of applicability for Darcy's law of saturated flow through porous media. Irmay (in Chap. 5 of Bear et al., 1968) mentions the existence of a *minimum (or initial) gradient* J_0 below which there is very little flow. For clay soils, J_0 may exceed 30. Irmay attributes this phenomenon to the rheological non-Newtonian behavior of the water." Bear goes on to discuss other explanations of non-Darcian behavior as put forward by other authors (Bolt and Groenevelt, 1969; Kutilek, 1969; Low, 1961; Swartzendruber, 1962).

Von Engelhardt and Tunn (1955) have compared the stagnant nature of connate waters to that of residual oil. "Connate waters (Haftwasser) adhere to the pore walls and do not take part in the fluid movement. They may be regarded in the same light as the residual oil that cannot be displaced from the pore spaces by the prevailing pressure gradients. The entire pore volume is therefore not available for fluid movement."

It is not clear at what pore-size ranges and under what conditions surface effects change liquid properties and Darcy's law no longer applies. Certainly, there is some pore size small enough that all pore water is held sufficiently close to mineral surfaces to be affected by the surface forces. When such a situation occurs, the flow will not be Newtonian, and Darcy's law will not apply.

Even if Eqs. 1 and 2 were rigorously applicable to a repository host rock, researchers would have difficulty determining the values of parameters such as hydraulic conductivity. Techniques commonly used for measuring hydraulic conductivity in the laboratory, in the field, or with inverse modeling techniques were developed for relatively permeable water-bearing granular media and are of uncertain applicability in repository studies.

Hydraulic conductivity values that have been determined from laboratory tests may be unreliable indicators of actual field conditions. Such determinations are made on small samples that, of necessity, have been disturbed. Flow in samples is one dimensional, whereas flow in the field usually is multidimensional. As Gloyna and Reynolds (1961) pointed out, "... fractures were apparently caused by the relaxation of stress which occurs when underground samples are removed from a compressed formation." Furthermore, laboratory determinations of hydraulic conductivity on low-permeability repository materials devoid of macrofractures and microfractures involve impractically long periods of time and measurements of exceedingly small flow rates. Therefore large pressure gradients across samples have been used to increase flow rates and to shorten the time required for experiments. Because the resulting flow may be non-Darcian, the so-called hydraulic conductivity values measured at the high pressure gradients may not reflect the properties of the rock under the lower natural hydraulic head gradients in the field.

If a threshold gradient is required to move water in certain materials, as some authors suggest, normal field gradients will not move water against the immobilizing interfacial forces except by molecular diffusion. Thus such dense rocks may contain fluids over the time frames now being considered for repository design with essentially zero loss by Darcian transport. Even if significant molecular transport of water occurs along grain boundaries or along openings of comparable size in these very dense materials, such transport is not the advective transport mechanism described by Eq. 2. In addition, the concept of hydrodynamic dispersion as employed in Eq. 2 does not seem readily applicable under such conditions. Thus, even if Eq. 1 should predict water transport under such conditions, radionuclide transport must be far less than that predicted by Eq. 2 using the normally determined coefficients.

On the other hand, Darcy's law does not hold for non-laminar flow in large fractures. Parameters measured using small laboratory samples do not include the effects of larger-scale geologic features such as joints and faults. Such features could greatly influence groundwater and solute transport rates. Consequently, the use of such laboratory data not only can overestimate water and solute transport rates in unfractured host rocks, but it can seriously underestimate the transport rates in fractured rocks.

A Reasonable Approach to Repository Design

Careful studies are being applied to the identification and characterization of suitable repository sites and to the preparation of repository designs. However, the final proving and testing of a nuclear waste repository must take place by underground testing of the in situ rock before and during the actual construction. Also there is a need to supplement the commendable simulation technology with some simple testing of the actual canister emplacement boreholes as the mined repository is developed.

Before proceeding, we should note that we are assuming that at least small volumes of rock devoid of both macrofractures and microfractures are present in the field. Gloyna and Reynolds (1961) present evidence that, at least for salt, this is likely.

Suppose a given canister emplacement borehole in the mined repository is tested at a large, but nondestructive, pressure. Assume further that analysis of the test results yields a hydraulic conductivity of less than 10^{-10} cm/s. Then it is likely that no significant fluid-transmitting fractures intersect that borehole. Furthermore, the flow regime could well be non-Darcian. Therefore it could reasonably be concluded that transport from that canister emplacement borehole would be only by molecular diffusion over the 10,000-year design period now specified. This conclusion would be reinforced if the safety factors introduced by other natural and engineered multiple barriers also were taken into account. On the other hand, if the test of the given borehole indicates the presence of either open fractures or Darcian flow, or both, that borehole should not be used for emplacement of nuclear wastes without engineering modification or sealing.

Methods of in situ testing and analysis and criteria for acceptance or rejection of canister emplacement boreholes in the mined repository should be developed. It may be necessary to test the holes in conjunction with heater tests and thermal design of canisters or to use other means to account for temperature effects on the hydrologic properties.

Conclusions

Siting of geologic nuclear waste storage repositories will be in rocks that are devoid of open macrofractures or microfractures. A fractured rock will be chosen as a host only if the fractures have been made essentially impermeable by the deposition of secondary minerals that will remain stable under repository conditions. Therefore site selection and design will depend on the transport of water and radionuclides through very dense, essentially impermeable materials. It is our concern that the extrapolation of normal hydrologic behavior, parameters, and equations to these materials that are characteristic of nuclear waste storage repositories may be leading to serious underestimation of their containment capabilities.

There is a danger in applying Darcy's law to dense unfractured rocks where transport may be non-Darcian and by molecular diffusion along grain boundaries. Application of Darcy's law has been justified as being a conservative approach that guarantees repository safety. However, it has confused the issue of hydrologic design of nuclear waste repositories, and it has inhibited the development of technically optimal plans for repository site identification and testing.

If it is determined that fluid transport in very dense unfractured candidate host rocks is non-Darcian and by molecular diffusion only, then useful methods of testing become apparent. In situ pressure testing of canister waste emplacement boreholes in the mined repository can verify the absence of fracture flow and the absence of Darcian flow. Such evidence, coupled with conservative choice of geologic environment and conservative engineering design, can guarantee waste isolation for the geologically short time frames now being considered.

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Field Testing at the Climax Stock on the Nevada Test Site: Spent Fuel Test and Radionuclide Migration Experiments

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~~Lawrence Livermore National Laboratory (LLNL) is~~ conducting ~~two field~~ tests in the Climax Stock are being conducted **~~at the Nevada Test Site in support of the Nevada Nuclear Waste Storage Investigations (NNWSI).~~ The Climax Stock, a granitic intrusive, has been administratively excluded from consideration as a full-scale repository site. However, it provides a readily available facility for field testing with high-level radioactive materials at a depth (420 m) approaching that of a repository.**

The major test activity in the 1980 fiscal year has been initiation of the Spent Fuel Test-Climax (SFT-C). This test, which was authorized in June 1978, is designed to evaluate the generic feasibility of geologic storage and retrievability of commercial power reactor spent fuel assemblies in a granitic medium. In addition, the test is configured and instrumented to provide thermal and thermomechanical response data that will be relevant to the design of a repository in hard crystalline rock.

The other field activity in the Climax Stock is a radionuclide migration test. It combines a series of field and laboratory migration experiments with the use of existing hydrologic models for pretest predictions and data interpretation. Goals of this project are to develop (1) field measurement techniques for radionuclide migration studies in a hydrologic regime where the controlling mechanism is fracture permeability; (2) field test data on radionuclide migration; and (3) a comparison of laboratory- and field-measured retardation factors.

This radionuclide migration test, which was authorized in the middle of the 1980 fiscal year, is in the preliminary design phase. The detailed program plan was prepared and subjected to formal peer review in August. In September/October researchers conducted preliminary flow tests with water in selected near-vertical fractures intersected by small horizontal boreholes. These tests were needed to establish the range of pressures, flow rates, and other operating parameters to be used in conducting the nuclide migration tests. 21 ref., 14 fig. 1 tab.

Introduction

Lawrence Livermore National Laboratory (LLNL) is conducting two field tests in the Climax Stock at the Department of Energy's (DOE) Nevada Test Site (NTS). The Climax Stock, a granitic intrusive, has been administratively excluded from consideration as a full-scale repository site. However, it provides a readily available facility for field test activities using highly radioactive materials at a depth (420 m) approaching that envisioned for a repository.

The major test activity, the Spent Fuel Test-Climax (SFT-C), is a full-scale test of the retrievable geologic storage of spent fuel assemblies from a commercial nuclear power plant. This test was authorized in June 1978, initiated in May 1980, and is planned to continue for three to five years. After the test is completed, researchers will remove the fuel assemblies for subsequent disposition.

The other field activity in the Climax Stock is a radionuclide migration test designed to develop field measurement techniques for studying nuclide migration in a hydrologic regime dominated by fracture flow. This test is designed to provide data for comparison of laboratory- and field-measured retardation factors for nuclides of interest in nuclear waste management.

Spent Fuel Test-Climax

Objectives. There are a number of reasons for carrying out a field test of spent fuel storage. One reason is that the public does not widely understand that handling spent fuel is a straightforward engineering problem that involves off-the-shelf technology; therefore a field test with spent fuel has educational value. In addition, technical benefits accrue from generic tests such as this one, which uses high-level radioactive waste. Even though much information can be gained from laboratory measurements and computer modeling, unpredicted negative synergistic effects that develop during storage are best examined in an actual field test. Thus the technical purpose of the test is to provide early field measurements of the effects of high-level radioactive waste storage in a crystalline igneous rock.

The general objective of the SFT-C is to evaluate the feasibility of safe and reliable short-term storage and retrieval of spent fuel assemblies at a plausible repository depth in a typical granitic rock (Ramspott et al., 1979). A secondary general objective, when consistent with the objective defined in the previous sentence, is to obtain technical data to address the evaluation of granite as a medium for deep geologic disposal of high-level reactor waste and the design of a repository in granite.

The following two main technical objectives have been defined so that the general objectives can be achieved:

1. To simulate the effect of thousands of canisters of nuclear waste emplaced in geologic media by using a small number of spent fuel assemblies and electrical heaters
2. To evaluate the difference, if any, between the effect of an actual radioactive waste source and an electrical simulator on the test environment

Secondary technical objectives are:

1. To compare the magnitude of displacement and stress effects from mining with the effects of thermally induced displacement and stress that occur after the spent fuel is introduced
2. To document quantitatively the amount of heat (about one-third, according to calculations) removed by ventilation
3. To compare the response to thermal load of relatively more fractured and less fractured rock

The layout of the test facility (Fig. 1) was designed to incorporate two experiments that address the two main technical objectives stated above. Technical objective one is addressed by having a central linear array of spent fuel and electrical simulators flanked by parallel arrays of electrical resistance heaters. The power history of the parallel arrays was developed by using computer modeling (discussed later) to simulate the effects of thousands of canisters of spent fuel in the center of the can-

- RADIATION EFFECTS EXPERIMENT
  - THE EFFECTS ON GRANITE OF HEAT ALONE (ELECTRICAL SIMULATORS) ARE COMPARED WITH THE COMBINED EFFECTS OF HEAT AND RADIATION (SPENT FUEL)
- REPOSITORY RESPONSE EXPERIMENT
  - THE RESPONSE OF A GRANITE REPOSITORY TO WASTE IS INVESTIGATED IN A REPOSITORY MODEL CELL BY USING A SMALL NUMBER OF SPENT FUEL ELEMENTS AND ELECTRICAL HEATERS

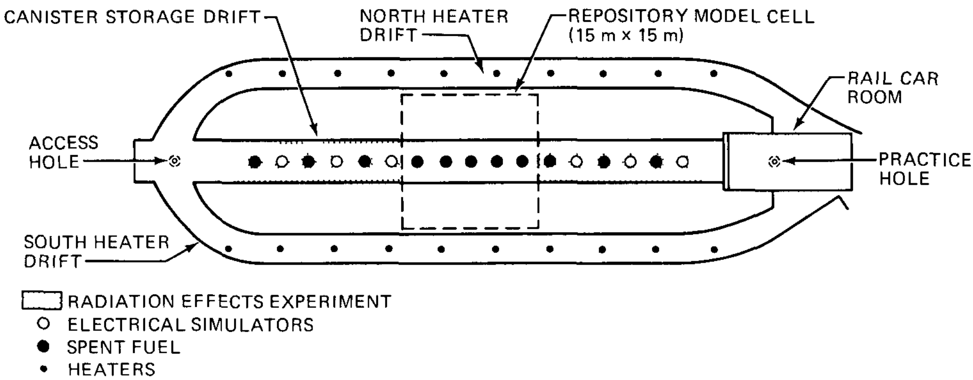

Figure 1 Plan view of Spent Fuel Test-Climax showing relation of test geometry to objectives.

ister drift. Technical objective two is addressed by alternating spent fuel with thermally identical simulators that use electrical resistance heating elements. Alternating these units on either end of the repository model cell exposes both relatively intact and highly sheared zones of rock to the two sources.

Geologic Setting and Site Characterization. The Climax Stock, which outcrops over an area of about four square kilometers in the northeast quarter of the NTS, is believed to expand conically downward to an area of about 100 square kilometers at a depth of several kilometers (Allingham and Zietz, 1961). Three major faults lie within one kilometer of the Stock. The north-trending Tippinip fault is west of the Stock; the northeast-trending Boundary fault forms the contact between the southeast border of the intrusive and the alluvium; and the north-trending Yucca fault is south of the Stock. The Yucca fault parallels the Boundary fault near the contact and may join the Boundary fault near the contact.

Climax Stock is composed of a granodiorite unit and a quartz monzonite unit. Both units contain quartz, potassium feldspar, plagioclase feldspar, biotite, and amphibole in varying proportions. In addition, the quartz monzonite contains pink alkali feldspar crystals up to 50 mm long in the matrix of 1- to 4-mm grains. The spent fuel test area is located in the quartz monzonite unit. Geologic investigations at the SFT-C took advantage of information obtained in the 1960's in support of nuclear weapons effects tests at the site. The previous studies defined three prominent joint sets (Maldonado, 1977) that are oriented N32°W-22°NE, N69°W-near vertical, and N35°E-near vertical.

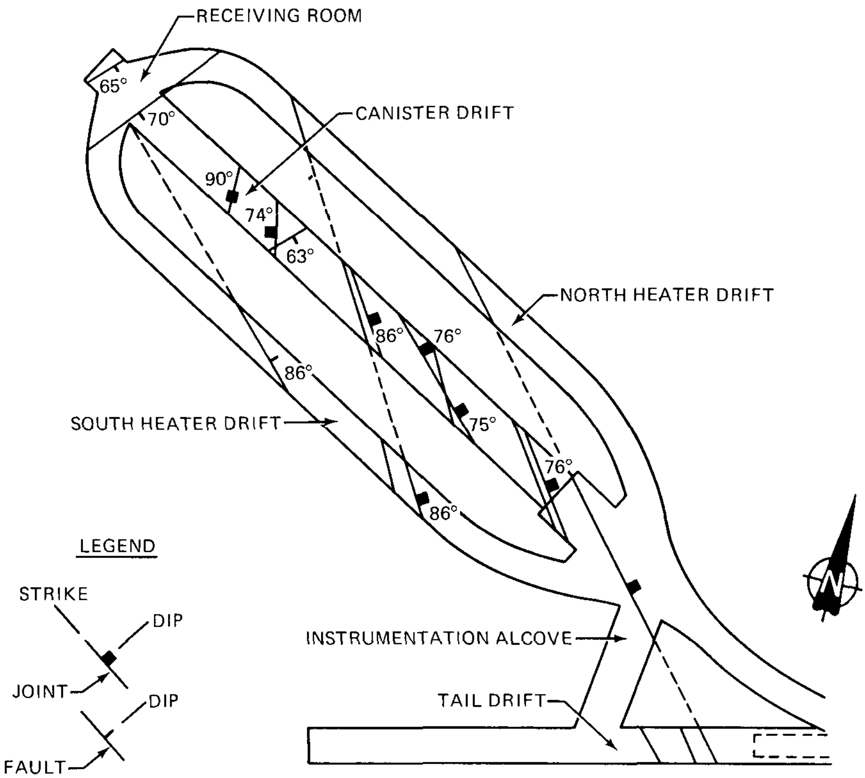

Figure 2 Locations of major geologic features within the Spent Fuel Test-Climax test area.

Four exploratory boreholes that were each 76 mm in diameter were cored through and beyond the proposed test area before starting construction of the SFT-C. These cores were used to verify the frequency and extent of fractures and the continuity of the Stock.

The ribs of all newly excavated drifts and the invert of the canister drift were mapped in detail. Data obtained are being reduced and analyzed (Wilder and Yow, 1981).

Initial indications are that the current mapping activities agree with previous studies. In addition to the three joint sets identified earlier, researchers have identified several zones of intense shearing and two faults in the test array (Fig. 2) (Wilder and Patrick, 1980).

Most instrumentation holes were cored, and record cores were obtained before drilling the storage holes in the canister drift. Heater holes in the heater drifts also were cored. More than 1000 m of core were obtained from holes that ranged from 38 mm to 152 mm in diameter. All cores will be logged. Results of the core-logging and fracture-mapping activities have at least four applications. First, they provide information for stability assessment and rock support selection. Second, they aid in siting instrumentation. Third, they are important in analyzing acquired data for effects of geology. Fourth, they are to be applied to thermomechanical computer modeling studies.

Laboratory studies indicate that the Climax Stock is typical of granitic

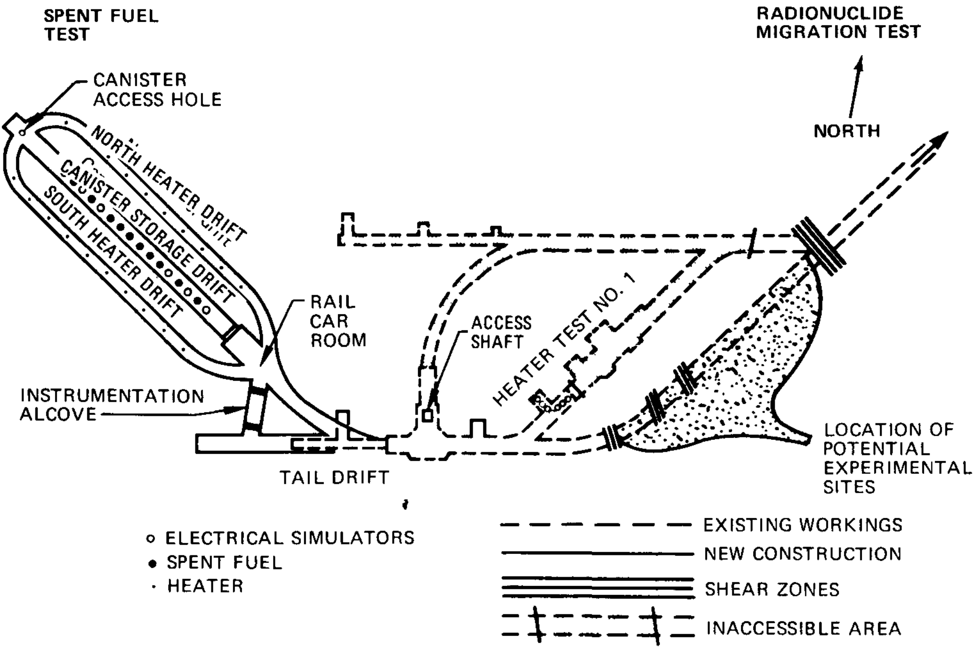

Figure 3 Layout of Spent Fuel Test-Climax and adjacent workings at the depth of 420 m.

rocks. In situ measurements indicate a modulus of 26 GPa and a Poisson's ratio of 0.246 (Heuze et al., 1981).

The in situ state of stress was determined using the U. S. Bureau of Mines (USBM) strain relief overcore technique. Resolving the principal stresses to the north, east, and vertical orientations gives a vertical stress of 7.92 MPa. This is somewhat less than the 10.9 MPa calculated to result from the weight of the overburden (Ellis and Magner, 1982). Results of a stress profile performed in one of the holes indicates that the measurements were made in a destressed block of ground. Although the stress magnitudes are suspect, the orientations and stress ratios are in agreement with other measurements at the Nevada Test Site (NTS).

Very little is known about the detailed hydrology of the Stock. A recent survey of the NTS hydrology indicates a regional water level 100 m to 200 m below the test facility (Murray, 1981). Measurements in a single borehole indicate a water level about 145 m below the SFT-C. Rock in the vicinity of the site apparently is unsaturated, but it is not dry. Several localized seeps occur in the excavations.

Test Layout. Access to the 420 m depth in the Climax Stock is provided by a two-compartment vertical shaft that is 2 m × 4 m. Originally developed in the 1960's to support nuclear weapons effects tests, this shaft and associated underground workings permitted development of the Spent Fuel Test facilities with a relatively minor rehabilitation effort.

Layout of the underground test array is shown in Fig. 3. Three parallel drifts were excavated in the area northwest of the vertical access shaft. The center canister storage drift is 4.6 m wide by 6.1 m high; this drift extends 65 m from a large room

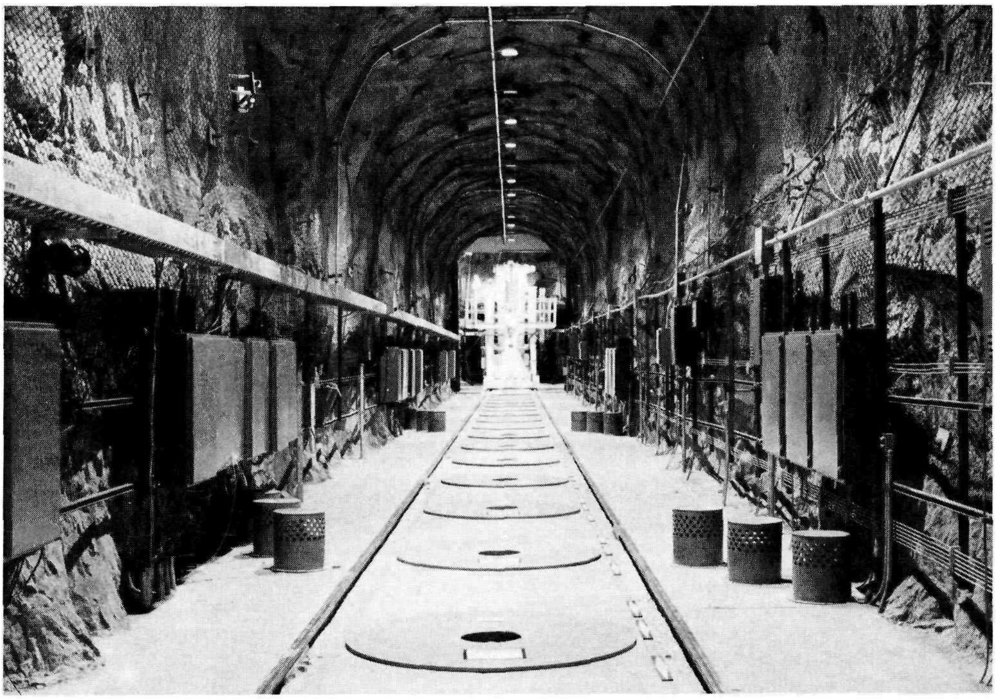

Figure 4 The Spent Fuel Test-Climax canister drift showing storage holes, some instrumentation, and, in the background, the underground transfer vehicle.

needed to construct and store the underground transfer vehicle to intersect with the bottom of a cased canister access and ventilation exhaust hole that is 485 mm in diameter (Fig. 4). The north and south heater drifts, 3.4 m × 3.4 m, parallel the storage drift and are spaced 10 m on center from it. These heater drifts provide a simulation of a small element of a room and pillar repository. They also allow for the installation of instrumentation to monitor the thermal and mechanical response of the test area and for the installation of electrical heaters that establish the appropriate thermal boundary conditions to simulate a large repository within the central drift and pillar region. The underground test area also includes an instrumentation enclosure and a tail drift extension sufficient to permit additional development without disturbance of the test measurements.

Supporting facilities on the surface include a headframe and hoist serving the access shaft, electrical substations and ventilation systems, equipment for transferring the spent fuel canisters underground, a data acquisition system trailer complex, and general mining support structures and equipment.

The test area is maintained under LLNL administrative control and is continuously monitored to confirm safe operating conditions. Provision for access to the test area by visitors is maintained; however, routine daily access to the underground facilities normally is not available.

Fuel Description. The spent fuel assemblies being stored in this test are Westinghouse, 15 × 15 PWR

(Pressurized Water Reactor) assemblies obtained from the Florida Power and Light Co. Turkey Point Unit #3 (a 666 MWe power plant).

Each fuel assembly contains 204 active fuel rods that are 10.22 mm in diameter and have an active fuel length of 3.66 m; the assembly cross section is 210 mm square. Initial fuel loading was 448 kg/assembly with a ^{235}U enrichment of 2.56%. The assemblies operated from 1974 to 1977 and were discharged with a fuel burnup of 28,000 MWD/MTU. At the time of emplacement into the storage array—2.5 years after discharge—the thermal power output was 1525 ± 40 W/assembly. Thermal power output was determined by calculations normalized to a calorimeter measurement made immediately before encapsulation of three assemblies from the selected matched set. Three assemblies were nondestructively examined before shipment to NTS, and a few fuel rods were removed for destructive evaluation. These three assemblies also were fitted with materials interaction test (MIT) capsules containing samples of various host rocks, water/brine, and other materials.

In addition to the 11 fuel assemblies that are emplaced for the test, another assembly is available for periodic rotation into the test array to verify retrieval capability.

Fuel Handling System. A canister handling system was designed, fabricated, and operated to transport the fuel canisters from the encapsulation facility to the test area, to lower the canisters underground, and to position them in vertical storage holes in the test array (Department of Energy, 1980). The fuel canisters are in a heavily shielded configuration at all times, and the sequence is reversible for retrieval at any time. This system, although not intended to be prototypical of repository support equipment, has demonstrated that the application of standard engineering practices can provide a safe and highly reliable system that performs all the required movements of fuel canisters with no significant radiation exposure to the operating personnel.

Each fuel assembly is encapsulated in a stainless steel canister that is 335 mm in diameter and 4.3 m long. The canister, which is welded closed and backfilled with helium at one atmosphere pressure, is suspended from a carbon steel shield plug that is 460 mm in diameter and 0.3 m thick. The shield plug is fitted with a stainless steel lifting knob.

All canister handling operations, except over-the-road transport, are remotely controlled to ensure minimum exposure to personnel. Special attention has been given to shielding, component redundancy, and accessibility to permit recovery from malfunctions.

The handling system has three basic subsystems: a surface transport vehicle (STV), a canister lowering system (CLS), and an underground transfer vehicle (UTV).

The STV is a three-meter-wide lowboy trailer coupled to a heavy-duty diesel-powered tractor. The trailer carries a transport cask that is made of steel, weighs 40 mt, is 1.15 m in diameter, and is 5.5 m long. The cask is trunnion-mounted on hydraulic jacks and is rotated from a near-horizontal transport position to a vertical loading/unloading attitude by hydraulic actuators. A hydraulically operated, hinged-top cover and an electrically powered sliding bottom gate permit loading or unloading through either end. Electrically powered hydraulic pumps and controls are remotely controlled through umbilicals. Manually installed locking pins secure the cask closures and the cask to the trailer during transport. The STV is driven under convoy approximately 75 km over NTS roads from the encapsulation facility to the test area.

The CLS, which transfers fuel canisters from the surface to the underground test area, consists of a steel-cased hole that is 415 m deep and 485 mm in diameter, a headframe over the access hole, and an electrically powered hoist that spools a special composite structural/electrical cable. The end of this cable is fitted with a remotely operated grapple and a grapple brake safety device. The access hole is fitted with shielding collars at both top and bottom. The top collar accepts the bottom gate of the surface transport cask so that a step-joint shielding configuration is provided as the fuel canister is transferred from the cask into the access hole. A similar geometry at the bottom of the access hole is employed as the fuel canister emerges from the hole and enters the transfer cask on the UTV.

The UTV is an electrically powered railcar that carries a 45 mt transfer cask and operates between the bottom of the access hole and each of the fuel canister storage holes in the 50-m-long test storage array. The UTV transfer cask, which has similar dimensions as the STV cask, has two-piece sliding gates—both top and bottom—that complete the shield configuration during the transfer operations and permit top loading (from the access hole) and bottom unloading (into the storage holes). The cask is raised into the shielding collar of the access hole during loading; once loaded, the cask is lowered to a mid-position while it is being moved to the location of the storage hole. When the UTV arrives at the proper storage hole, the cask is lowered into a shielding collar that is similar to the collar at the access hole (each storage hole has such a shielding collar) and the cask is unloaded. The UTV also is equipped with an on-board jib-mounted hoist and canister grapple that is used to lower the fuel canisters into the storage holes and to handle concrete-filled shield plugs that cover the fuel canisters in the final storage configuration.

All three of these subsystems are operated from a single control room that is located on the surface above the test array and adjacent to the access hole hoist. Control consoles, closed-circuit television monitors, remote radiation monitoring indicators, and test ventilation system controls are located in this control room. Canister emplacement and retrieval operations are performed by a three-man crew under detailed technical operating procedures.

Shipment from the encapsulation facility to the test area, transfer to the test level, and emplacement of a fuel canister in the storage array require one shift per canister.

Storage Configuration. Eleven spent fuel canisters and six electrical simulators are emplaced in vertical storage holes constructed in the floor of the center drift. The storage array consists of 17 identical holes spaced three meters apart and centered in a single linear row between the rails that support the UTV. The storage configuration is shown schematically in Fig. 5.

Storage holes were constructed using a full-face bit that was 610 mm in diameter and driven by a downhole percussion drilling machine. A carbon steel pipe-liner assembly was suspended in the hole and grouted at the top and bottom only; the interval opposite the active fuel length was left ungrouted. The liner, which is designed to be retrieved at the end of the test, is fitted with a machined conical seat to support the canister shield plug. Therefore the fuel canisters and simulators hang free from support pins below the shield plug and do not contact the liner assembly.

At the top of each storage hole is a steel-lined pit that is 1.4 m in diame-

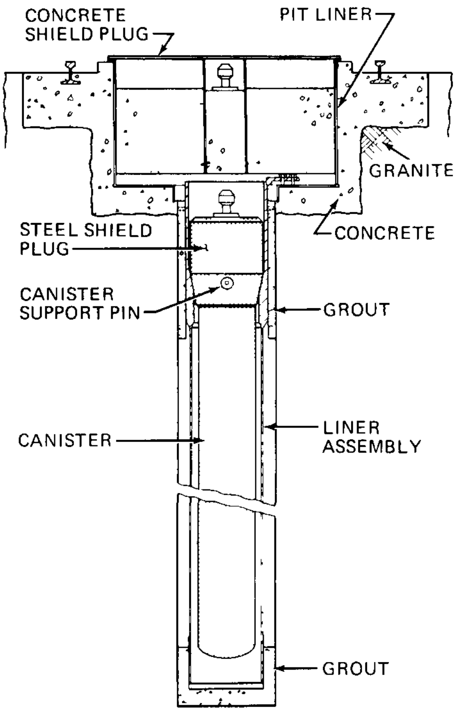

Figure 5 Cross section of the Spent Fuel Test-Climax canister storage configuration.

ter and 0.8 m deep. This pit provides a recess for the UTV transfer cask and completes the storage configuration when a steel-encased concrete shield plug is placed in the pit.

Each hole is equipped with gas sampling tubes and several of the holes also contain passive dosimeters to monitor the ionizing radiation exposure to the host rock.

Instrumentation and Data Acquisition. Detailed discussions of the rationale and criteria associated with instrumentation design, selection, procurement, installation, and calibration are presented elsewhere (Carlson et al., 1980; Brough and Patrick, 1982). A brief overview that draws on the cited references is presented here.

Numerous instruments were installed and monitored to obtain data necessary to address the technical objectives of the SFT-C. Technical measurements include temperature, stress, displacement, radiation dose in granite, and ventilation parameters. In addition, radiation safety monitors are operated continuously. Other instruments are used to monitor the environment in which the equipment operates and the reliability of the data acquisition system.

Both near-field and intermediate-field temperature measurements are employed at the SFT-C; Type K Chromel-Alumel thermocouples in magnesium-oxide-packed Inconel tubing are used. Near-field temperature measurements achieve three purposes: verification of canister power levels, comparison of heat transfer from spent fuel with that from electrical simulators, and thermal correction to other instruments (Brough and Patrick, 1982).

Each canister storage hole has 18 thermocouples associated with it (Figs. 6 and 7). Six of the thermocouples are located in contact with the spent fuel canister at two azimuthal and three axial positions. Another six are on the steel liners at similar locations. Three additional thermocouples are located in each of two holes 0.50 m and 0.66 m from the canister centerline.

Some intermediate-field thermocouples are closely associated with extensometers to facilitate temperature corrections to these instruments. Others measure temperature beneath the canister drift and laterally beyond the limits of the excavation (Fig. 8).

Air temperature and flow monitors are used to measure the temperature and humidity of the air at the inlet and exhaust of the test array. Data collected provide information on the amount of heat being removed by the ventilation system. Air flow is mea-

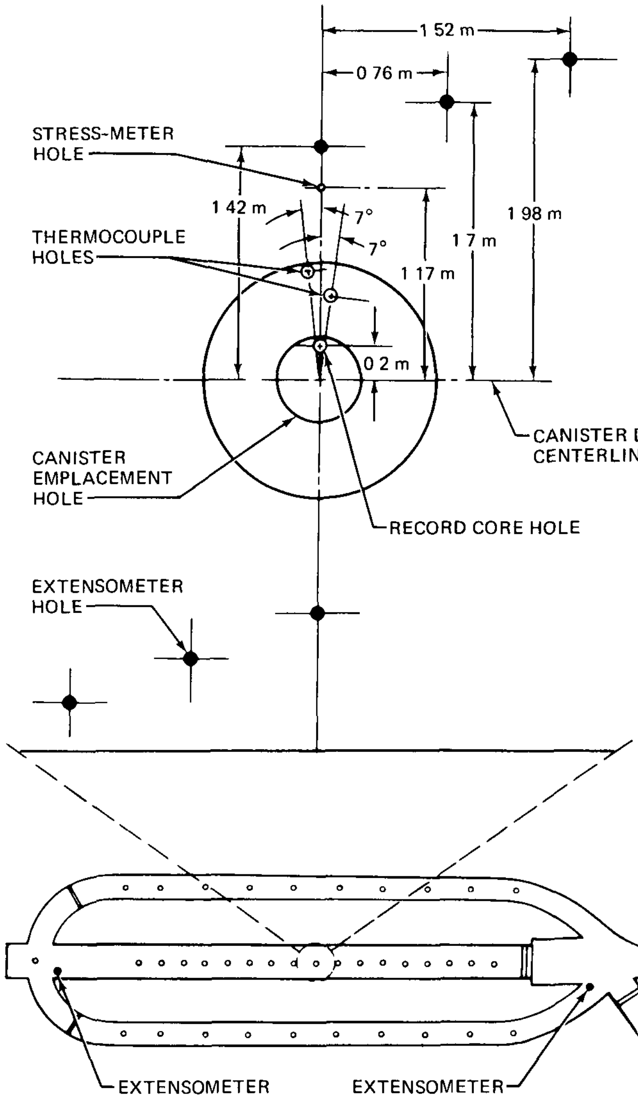

Figure 6 Plan view of extensometer, stress meter, and near-field thermal instruments at two heavily instrumented locations in the canister drift.

sured by a turbine flowmeter located at the surface in the ventilation ductwork.

Vibrating wire stressmeters are used to record changes in rock stress at several locations in the facility.

Twelve multiple anchor rod extensometers were installed in the pillars to measure relative displacements during the mining operation. All rod extensometers at the SFT-C employ hydraulic anchorage and use precision linear potentiometers as displacement transducers. These units were refurbished before the heated phase, and thermocouples were installed in paral-

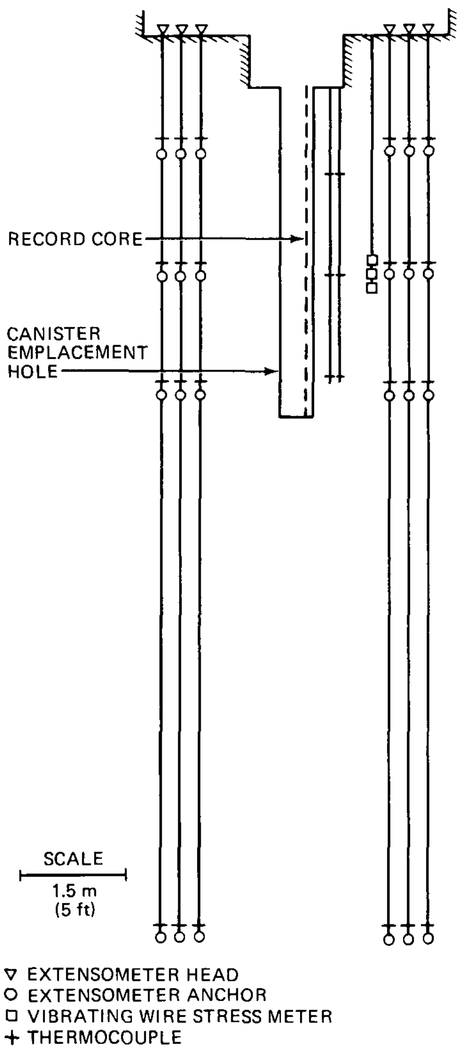

Figure 7 Projected vertical section of extensometer, stress meter, and thermal instruments at two heavily instrumented locations in the canister drift.

lel holes to permit thermal corrections to the mild steel rods.

The canister drift rod extensometers consist of 14 vertically oriented four-anchor units. Six are clustered around the CEH03 hole and six around the CEH09 hole; the remaining two are located at the far ends of the test array (Figs. 9 and 10). Several improvements were made to the rod extensometers deployed in the canister drift. First, a pressure regulating system was added to preclude rupture of the anchor system at elevated temperatures. Second, Superinvar rods were employed to reduce thermal expansion. Third, the assemblies were sealed and grouted in place to eliminate the formation of convection cells in the annular space. Closed-cell-foam rings were placed above and below each anchor, however, so that the anchors were not mechanically connected to each other by the grout column.

Convergence wire extensometers were deployed at five locations in each heater drift, at six locations in the canister drift, and at two locations passing through the pillars from one side of the facility to the other (Figs. 9 and 10). These units were designed and fabricated at LLNL to provide measurements of drift convergence and divergence. These convergence wire extensometers use a linear potentiometer as a transducer and have functioned reliably to date. Tape extensometer measurement points at the convergence wire anchors provide a mechanical check on the electronically recorded measurements.

Measurement of displacements across discrete geologic features is achieved by using fracture monitor systems designed and fabricated at LLNL. Three components of displacement are measured at each of seven locations. Linear potentiometers sense displacements along the strike, dip, and normal to the plane of seven zones of intense jointing.

Radiation dose to granite is measured with specially constructed dosimeters. Gamma radiation is measured with optical-grade lithium fluoride crystals; neutron radiation is measured with cobalt

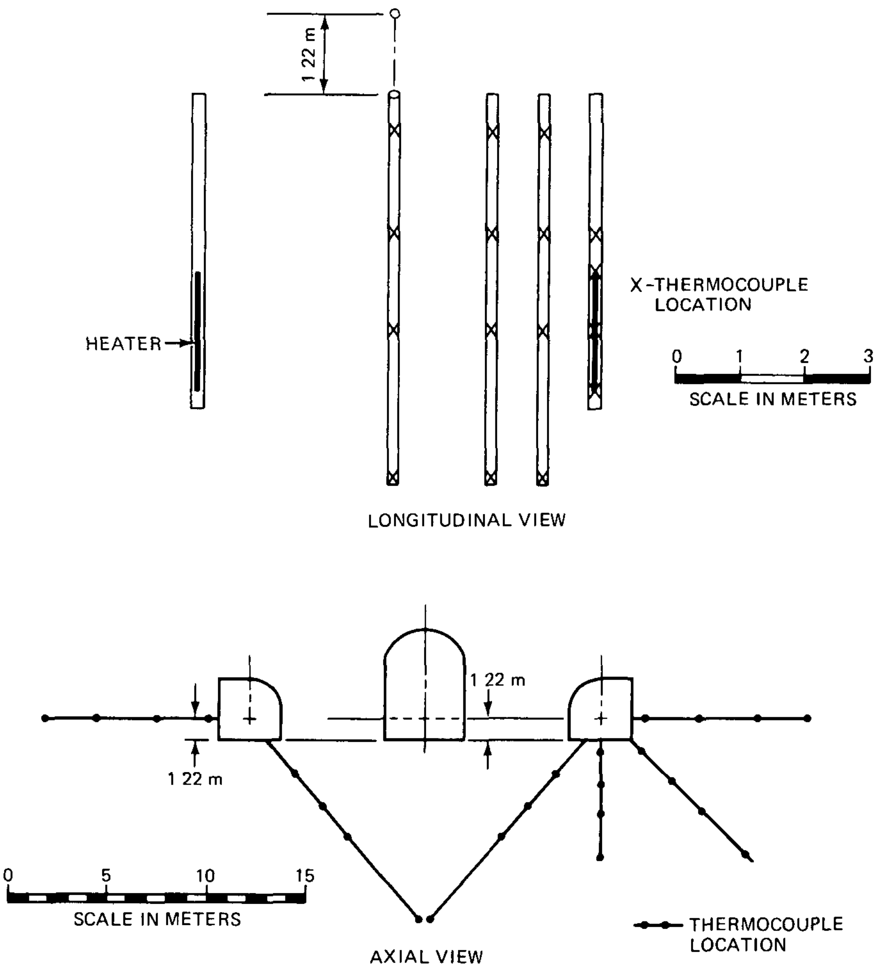

Figure 8 Intermediate-field thermocouple locations in the abutments and below the test array.

foils and the nickel content of the stainless steel parts of the dosimeter assembly. Dosimeters are located in guide tubes secured to the hole wall at five storage holes. One of the five holes having dosimeters contains an electrical simulator; the dosimeters in this hole serve as a control. Additional dosimeters are located in the near-field thermocouple holes 0.50 m and 0.66 m from the canister centerline at one location. Dosimeters are periodically removed and read, and the data are corrected for thermal annealing.

Basically, the data acquisition system is two Hewlett-Packard Series 1000 disc-based minicomputer systems configured for high-performance real-time systems applications. Each computer system is equipped with a 20-megabyte cartridge disc, a nine-track digital magnetic tape unit, an interactive display terminal, a high-resolution digital voltmeter, and six 80-channel scanners. In addition, one

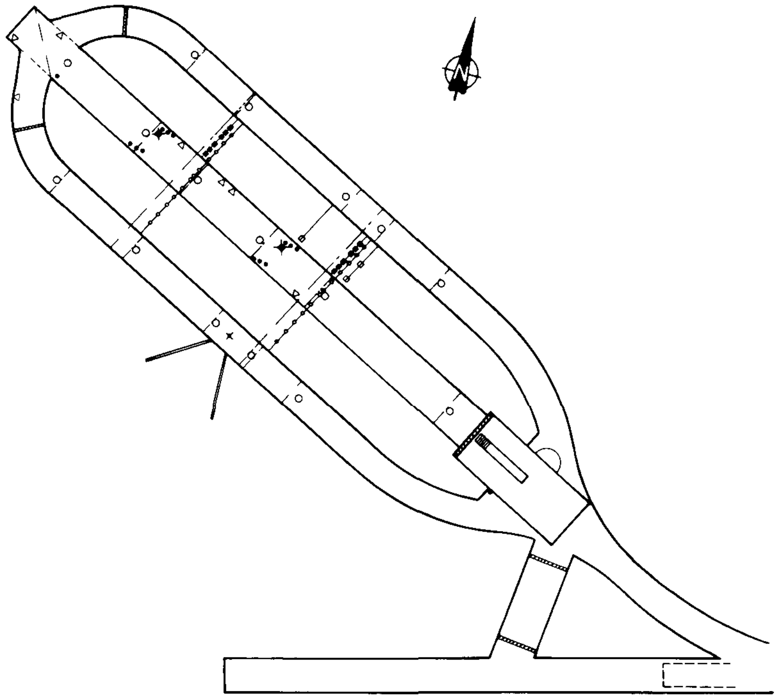

Figure 9 Plan view showing location of stress and displacement instrumentation in the Spent Fuel Test-Climax.

computer controls a line printer, an IRAD Stressmeter Datalogger, a multicolor graphics plotter, and TV status monitors.

Scanners and digital voltmeters are located underground where they digitize and transmit data to the main computer system that is housed in a trailer on the surface.

Two remotely located terminals are connected to the system by telephone modems. One of the terminals is located at the NTS communication center, which is manned 24 hours a day, and provides surveillance for abnormal conditions. The other terminal is located at Livermore and allows project personnel access to all data acquisition functions.

Software for the data acquisition system includes both Hewlett-Packard standard and special software as well as applications software developed by LLNL.

Uninterruptible power supplies (UPS) located underground and on the surface furnish support for the data acquisition system. These units pro-

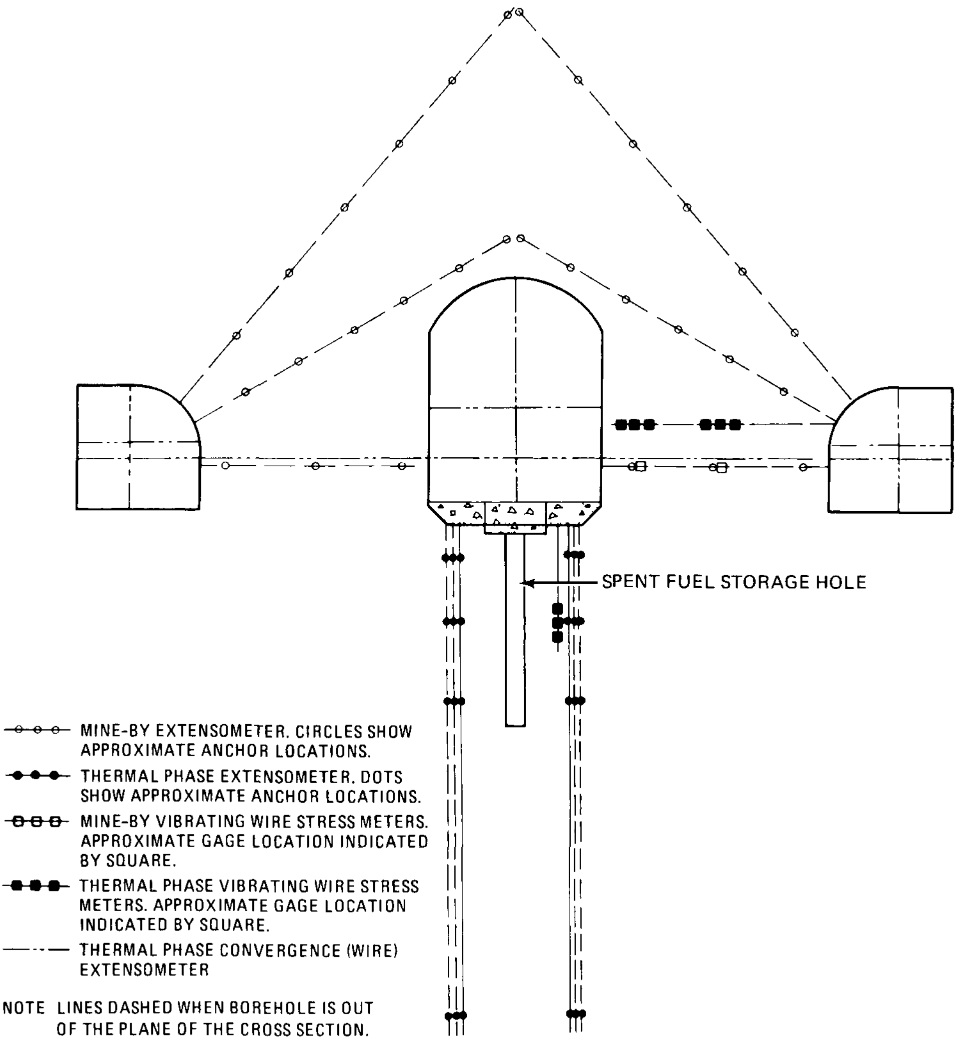

Figure 10 Projected vertical section showing location of stress and displacement instrumentation in the Spent Fuel Test-Climax.

vide isolation, regulation, and filtering of utility power and system backup power for up to 2.5 hours in the event of a commercial power outage.

The data acquisition system records about 0.3 million data points on magnetic tape each month. Copies of the magnetic tapes are shipped to Livermore for validation, reduction, and analysis.

Subroutines are now being developed to provide more accurate conversion of raw data than currently is performed by the field system. For example, thermal corrections to displacement data must be made using fairly complex algorithms. A data compression scheme is being developed to reduce the size of data files and to improve the data reduction and analysis process.

Modeling and Predictions. Computer codes were used to model three processes of concern in radioactive waste isolation: heat transport from thermal sources, thermomechanical response of the rock mass, and radiation dose to the rock.

Heat transport calculations were performed at several stages during the test. The level of sophistication of the models employed was based on the complexity of the problem geometry and on the level of accuracy required. Output from these calculations was used in safety assessments, instrument selection and location, and ventilation system design.

During the conceptual design, thermal modeling was used to establish the test geometry (Ramspott et al., 1979; Montan, 1980). These calculations indicated that a small test facility can simulate the near-field thermal response of a full-scale repository (8000 canisters in an array 600 m × 600 m) for early times after emplacement. When this scheme is used, errors in the center of the repository cell do not exceed 5% during the first 100 years.

Detailed pretest thermal calculations also were made (Montan, 1980; Montan and Patrick, 1981). These employed the multidimensional TRUMP code (Edwards, 1972). Both two- and three-dimensional modeling with conduction, convection, thermal radiation, and ventilation were employed. A significant result of these studies was the finding regarding the importance of thermal radiation. Many researchers treat underground openings as nonconductive voids and ignore thermal radiation. This leads to erroneous results in terms of temperature and, implicitly, thermal stress. Ventilation also has a significant effect on the thermal regime. In this three-to-five year test, about one-third of the heat will be removed by ventilation.

Another important thermal analysis involved calculating the cladding temperature of the fuel pins. This calculation indicated a peak cladding temperature of 290°C; the maximum allowable cladding temperature is 380°C.

Thermomechanical response of the rock mass also was calculated at several stages of the test. Most modeling employed the ADINA structural analysis computer code and its companion heat transport code ADINAT. The first phases of modeling used laboratory material properties and addressed the mechanical response of the rock mass to the mining of the canister drift. Maximum displacements on the order of two millimeters were calculated to occur (Ramspott et al., 1979; Butkovich, 1980).

The mining response calculations have been repeated using field-measured material properties and in situ state of stress (Butkovich, 1981). Other calculations that use a discrete joint model known as JPLAXD also have been performed (Heuze, Butkovich, and Peterson, 1980). The presence of discrete joints significantly affects the magnitude and orientation of displacements and stresses.

Finally, a set of calculations using best-available material properties has been performed to model mechanical response to mining and response through the first five years of heating the rock mass (Butkovich, 1981). These calculations show the importance of using the correct modulus of the rock mass as well as the significance of a zone of blast-damaged rock 0.5 m thick surrounding each underground opening. These calculations also considered thermal radiation and ventilation (Butkovich and Montan, 1980).

As indicated earlier, an objective of the test is to document the effects of ionizing radiation on the host rock. Radiation dose to the rock has been calculated in support of these studies. Radiation dose calculations were performed using the MORSE-L Monte Carlo radiation transport code (Wilcox and Van Konynenburg, 1981; Wilcox, 1972). Both gamma and neutron radiation are modeled. Calculated dose over the center 2.44 m of the canisters is nearly constant. The initial maximum dose rate to the granite is calculated to be about 9×10^3 rads/h; this will yield an integrated dose of 220 megarads during the proposed five-year test duration (Wilcox and Van Konynenburg, 1981).

The exponential component of attenuation is dominant in the first 0.3 m of granite, which is the region in which dose measurements are being made. This fact is reflected in the linear nature of the dose rate vs. distance semilog plots.

Significant Results and Status. As of June 1981 the test had been in full operation for slightly longer than one year. At this time several results can be reported. The spent fuel assemblies were transported from the power plant to NTS, encapsulated, and emplaced in the storage array without significant incident. The fuel handling system operated successfully in all respects. After eight months in storage, one fuel canister was retrieved and exchanged for another; thus a demonstrated retrieval capability was established.

Thermal distribution within the test array has developed in very good agreement with the calculated predictions, and confidence in the adequacy of the thermal modeling capability (Fig. 11) has been established. Peak fuel canister surface temperatures of ~145°C were attained after about three months in storage. Peak storage temperatures for the hole wall were observed at ~80°C after about eight months. Temperatures are now declining slowly (~1°C/month). No anomalous thermal effects that would indicate degradation of the near-field host rock have been observed. All thermal instrumentation is operating reliably.

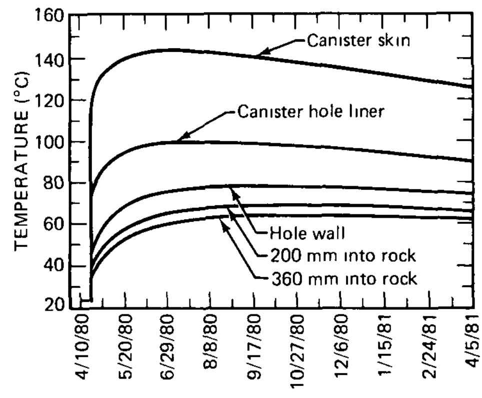

Figure 11 Plot of temperature vs. time at selected near-field locations for the first year of the test (note that all locations have attained peaks and are cooling slowly).

Preliminary evaluation of the rock mass displacement measurements indicates that all displacements are small (< 2 mm). The thermomechanical calculations employ linear-elastic continuum models that do not adequately describe the fractured nature of the host rock. Nevertheless, the calculated and observed displacements associated with thermomechanical response of the rock are in reasonable agreement. No quantitative measurements or visual observations imply any significant concern for the structural stability of the test area excavations. Some malfunctions of stress and displacement instrumentation have occurred and remedial work is in progress.

Interest in the progress of this test is evidenced by the level of public information activity. During the past year more than 130 tours have been conducted for more than 1700 representatives of technical, governmental, and public interest groups from throughout the United States and 22 foreign countries. Many technical reports—some of which appear as references in this article—have been published on specific aspects of the test.

At this time (June 1981) the test is continuing consistent with plans. The spent fuel assemblies have decayed to a power level of < 1000 W/assembly. Data acquisition and analysis are a continuing routine activity.

Future Plans. Planned duration of this test is three to five years. Current plans include several periodic fuel canister exchanges to confirm a continuing retrieval capability. At the termination of the test, researchers will retrieve all the fuel canisters and the canisters will be returned to the encapsulation facility for subsequent disposition. Instrumentation in the test facility will continue to operate for several months after all the fuel canisters have been retrieved so that the cool-down phase of the thermal cycle will be monitored. The storage hole liners will be removed and post-test core samples of the hole walls will be recovered for analysis of the effects of the thermal and radiation environment on the rock.

At that point, several options are available. The test array could be reconfigured to provide a test bed for more advanced multiple-barrier waste packages; additional testing (perhaps at significantly higher power levels) could be done using electrically heated thermal sources; or the facility could be decommissioned. Programmatic decisions among these options will be made in the next few years.

Radionuclide Migration-Climax

Objectives. Information on fluid flow and radionuclide transport through fractures is needed to evaluate the potential of granite as a host rock for a nuclear waste repository. Previous research on rock-waste-water interactions emphasized laboratory studies of sorption and radionuclide migration. Field tests are now needed to determine whether laboratory studies accurately reflect in situ conditions.

Field studies briefly described in this document are designed to meet three objectives:

1. To study radionuclide migration in fractured granite
2. To compare retardation factors measured in the field with values measured in the laboratory
3. To develop a reliable in situ retardation test that can be used at any repository site in fractured rock

To meet these objectives, we will use existing hydrologic models for pretest predictions and data interpretation.

Technical Approach. The field radionuclide migration experiments will use two boreholes that are approximately two meters apart and are drilled to intersect a high-angle fracture (see Fig. 12). Fractures selected for the migration experiments must be:

1. Greater than three meters from the drift wall to avoid the fracture zone induced by mining
2. Greater than three meters away from the drift along the fracture plane to avoid fluid flowing out of the fracture into the drift
3. A single fracture not associated with a major shear zone

Straddle packers will be set to isolate the fracture connecting the upper and lower boreholes.

An experiment will consist of:

1. Establishing a steady-state flow along the fracture by injecting into the upper borehole and monitoring the flow out of the lower borehole
2. Injecting both sorbing and non-sorbing radionuclides into the flow at the upper borehole
3. Monitoring for the presence of radionuclides by using on-line sensors at the outlet flow
4. Collecting samples for laboratory analyses

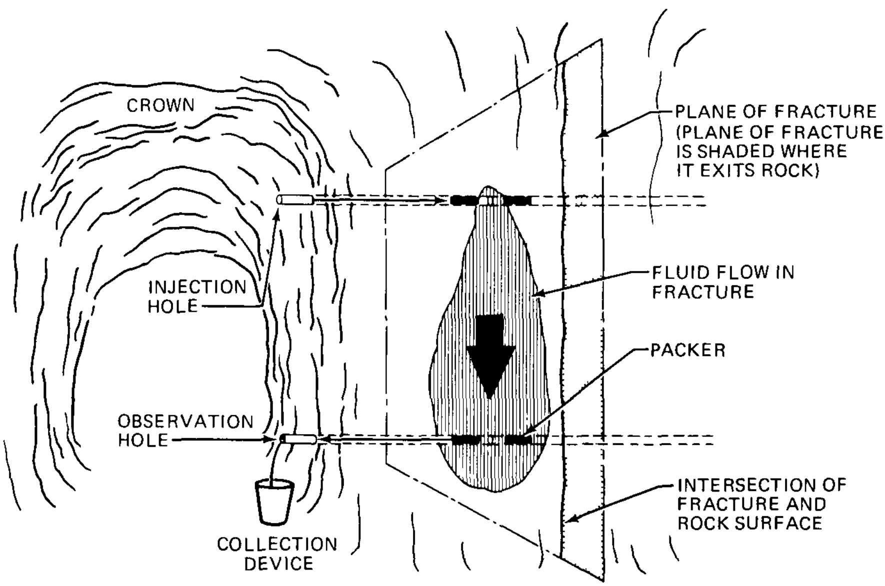

Figure 12 Schematic of radionuclide injection and collection scheme showing borehole access to high-angle fractures.

Retardation factors will be calculated by comparing the arrival times of the sorbing nuclides with the arrival times of the nonsorbing nuclides. More than one experiment will be conducted in the same fracture. Following the migration experiments, researchers will drill a network of cores between the two holes to determine the location of radionuclides that may have precipitated or have very high retardation factors. Length of time for each experiment, number of fractures used, and the sequence of experiments will depend on the time available and the results from each previous experiment. Designed to be limited in scope, these field tests will not give a regional picture of radionuclide migration potential in granite; however, they will provide scaling factors of several orders of magnitude over laboratory tests.

Hydrological Investigations. Initial field activities have concentrated on hydrological investigations to evaluate the flow characteristics of different fractures and to determine whether fractures in the Climax Stock are suitable for the migration experiments as planned. A critical question was whether we could successfully isolate an individual fracture between two boreholes and establish flow along that fracture such that most of the injected fluid would be recovered from the outlet hole.

Equipment used to conduct the initial hydraulic tests was designed to monitor and control a maximum range of pressures and flow rates. The maximum injection pressure of 300 psig is limited primarily by the packer inflation pressure. Maximum flow rate varies from 4 l/min at 10 psig to 0.8 l/min at pressures greater than 100 psig. Because access to the drift is limited by the size of the shaft, we designed a system composed of modules that could be lowered individually down the shaft. The modules are mounted on a rail cart that is easily positioned along the drift at any of the experimental sites (Fig. 13).

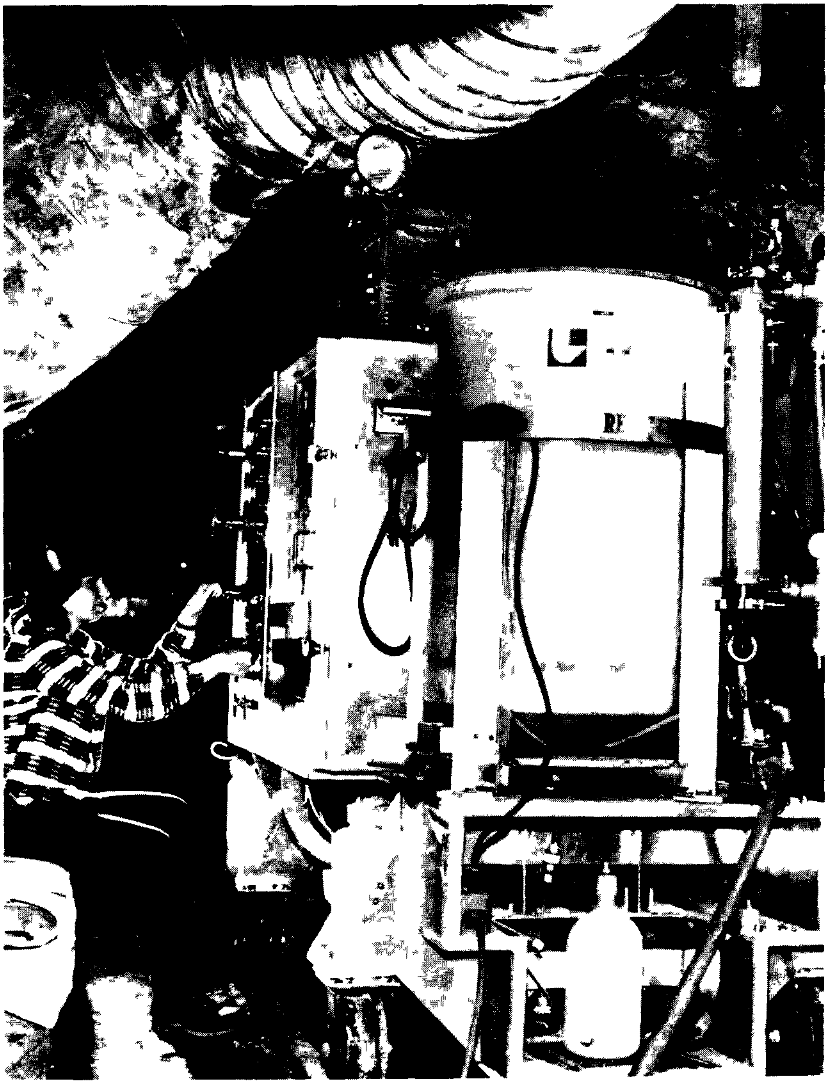

Figure 13 Rail-mounted instrumentation cart that supports the fracture characterization testing for the radionuclide migration experiments.

The five modules consist of:

1. A manually operated fluid injection manifold and pressure controller.
2. Pressure vessels and load cells for measuring the weight of fluid injected.
3. A clean water reservoir connected to an ultrafilter unit capable of removing particles < 0.05 μm.
4. A lower packer manifold and load cell for measuring the weight of the collected outlet flow.
5. A data acquisition system that continually records pressures, weights, and time.

Inflatable straddle packers were modified to allow for pressure transmission lines from both the packed-off interval and the borehole section at the far side of the packer. Both the hydraulic test equipment and the packers were tested in the laboratory before use in the field.

Three sets of borehole pairs have been drilled in the Piledriver drift. Each hole was logged visually with an optical borescope to locate high-angle fractures suitable for testing based on the selection criteria stated in the Technical Approach. Of the ten fractures tested to date, one fracture would not take water at pressures up to 200 psi for 24 hours. Several fractures were either so permeable they exceeded the pumping capacity of the equipment or else failed to show a connection between the two boreholes. In two fractures we were able to establish a circulating system with up to 95% of the injected water recovered. Circulating and noncirculating constant rate injection tests were conducted to determine the flow characteristics of these fractures. Analysis of the test data using a radial flow model gave fracture transmissivities of 5×10^{-4} m^2/d and 1.4×10^{-3} m^2/d. Fracture apertures were estimated at 20 μm and 30 μm, respectively.

The ability to establish circulating flow in the two fractures, combined with recovery of up to 95% of the injected water, gives us confidence that the migration experiments are feasible. Several fractures in the existing boreholes remain to be tested. To expand the number of candidate fractures, we will drill, log, and test at least one more set of boreholes before the final site selection for the first radionuclide migration experiment.

Groundwater Characterization. Concurrent with the hydraulic testing activities is a study of the Climax groundwater chemistry. As mentioned earlier, the 420 m level is above the local water table; however, groundwater seeps are found in several locations. An apparatus was installed in the Piledriver drift against the crown of the drift to collect water dripping from the intersection of two major fractures (Fig. 14). The crown was shaped to fit an O-ring that seals the apparatus against the crown and prevents air from contacting the water sample once the apparatus has been flushed with high-purity nitrogen. By isolating the sample from the atmosphere, we will determine pH, Eh, dissolved oxygen, and distribution of chemical species in the water as it flows through the fractures. Our first results, obtained by using an electrode chamber attached to the collection vessel to give on-line measurements, showed the pH and Eh to be relatively constant. Unfortunately the range of values for dissolved oxygen suggested a leak in the apparatus; therefore the apparatus has been reset recently and we are continuing to sample. Analytical results from the first two samples show water that is high in dissolved solids (~1900 mg/l) and rich in sodium, calcium, chlorine, and sulfate (SO_4) (Table 1). So far, the unusually high amount of sulfate cannot be explained by the geochemistry of the local rock.

Tritium analyses of the two samples show a tritium level of 135 pCi/ml, which is far above normal background (0.3 pCi/ml). Similar tritium levels have been found in the shaft water, the seeps in the Spent Fuel Test drift, and in a 200-m-deep NX hole drilled in the floor of the drift. We believe the tritium has migrated from the Piledriver nuclear test (migration probably occurred at the time of the test). If this is the case, the water probably is contaminated with ^{36}Cl and ^{14}C and will be difficult to age-date.

The anomalously high level of uranium (~1.8 ppm) found by using

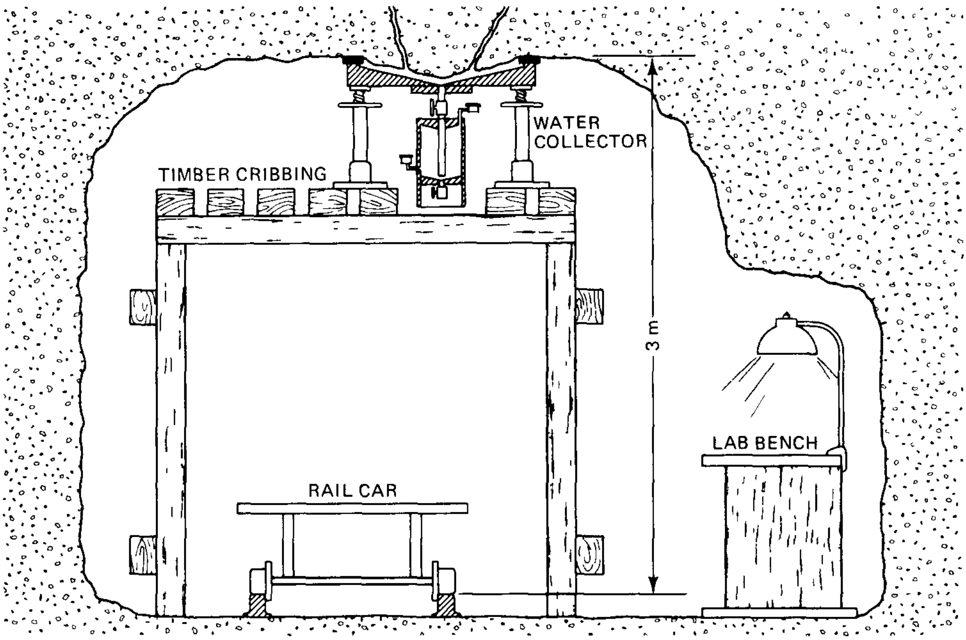

Figure 14 Schematic of the special water collection and analysis apparatus used in the Climax groundwater characterization studies.

Table 1 Climax Groundwater Chemistry—Preliminary Results

Constituent	Concentration	Constituent	Concentration
Na	243 mg/l	SO_4^{-2}	1050 mg/l
Ca	270 mg/l	S^{-2}	< 0.05 mg/l
K	2.2 mg/l	As	< 0.1 mg/l
Si	7.8 mg/l	Fe	0.01 mg/l
Al	0.03 mg/l	B	1.0 mg/l
Cl	70 mg/l		
Mg	1.1 mg/l	Total Dissolved Solids	~1900 mg/l
PO_4^{-3}	0.5 mg/l		
F^-	1.4 mg/l	pH	7.7
U	1.8 mg/l	Eh	0.46 V
Mo	0.7 mg/l	Dissolved O_2	~0.3 ppm
Sr	5.0 mg/l	Alkalinity	146 ppm HCO_3^-
		Conductivity	2200 μmhos

both the inductively coupled plasma (ICP) spectrophotometer and site-selection fluorescence spectroscopy techniques were confirmed by a complete isotopic spectral analysis. A $^{234}U/^{238}U$ activity ratio of 1.115 ± 0.012 is within the normal range for groundwaters. The $^{235}U/^{238}U$ activity ratio of 0.04455 ± 0.00167 is indistinguishable from the natural ratio of 0.04607.

To date, we cannot explain why the Climax groundwater contains so much uranium, even if it is natural. Fractures in oxidized, weathered, granite bodies frequently contain uraniferous phosphates (varieties of autinite and torbernite). For the groundwater to maintain this high value of uranium, the fractures immediately adjacent to the drift ceiling must contain these uraniferous phosphates or other minerals; otherwise the uranium would be diluted by more normal water containing uranium in the parts per billion (ppb) range. The other possibility is that the tuffs overlying the Climax Stock were a source of uranium to waters that ultimately became Climax groundwater.

Interpretation of the complex chemistry of the Climax groundwater will require the use of LLNL's geochemical modeling codes EQ3 and EQ6 to determine the equilibrium state and chemical evolution of the groundwater. Code EQ3 will be used to determine the solute species distribution in the samples and to calculate the saturation state of the samples with respect to the Climax primary and secondary minerals. Code EQ6 will be used to make reaction-path models of groundwater evolution to compare with the groundwater analyses.

Plans and Schedule. This project began in the spring of 1980. A program plan was developed that describes in detail the scheduling of the various tasks and how they interrelate (Isherwood et al., 1980). Hydrologic investigations were initiated in the fall of 1980 and will continue as long as is needed to characterize the fractures. On the basis of the results of the initial hydrologic tests, we will design the specialized injection-collection equipment needed to conduct and monitor the radionuclide migration experiments. In 1982 we will fabricate, install and test the equipment, and begin the first set of experiments. Continuation of the laboratory sorption experiments will be concurrent with the field activities in 1982 and 1983. We plan to complete the migration studies in 1983. Cores will be drilled between the inlet and outlet holes to determine whether radionuclides remain in the fracture. Analysis of these cores will take place in late 1983. In 1984 the final report will be prepared, reviewed, and released. Preliminary plans call for initially using tritium, ^{85}Sr, ^{95m}Tc, ^{131}I, and ^{137}Cs. If the first series of migration experiments is successful, a second series using actinides will be proposed.

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The AEGIS Geologic Simulation Model

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The Geologic Simulation Model (GSM) is used by the AEGIS (Assessment of Effectiveness of Geologic Isolation Systems) program at the Pacific Northwest Laboratory to simulate the dynamic geology and hydrology of a geologic nuclear waste repository site over a million-year period following repository closure. The GSM helps to organize geologic/hydrologic data; to focus attention on active natural processes by requiring their simulation; and, through interactive simulation and calibration, to reduce subjective evaluations of the geologic system.

During each computer run, the GSM produces a million-year geologic history that is possible for the region and the repository site. In addition, the GSM records in permanent history files everything that occurred during that time span. Statistical analyses of data in the history files of several hundred simulations are used to classify typical evolutionary paths, to establish the probabilities associated with deviations from the typical paths, and to determine which types of perturbations of the geologic/hydrologic system, if any, are most likely to occur. These simulations will be evaluated by geologists familiar with the repository region to determine validity of the results. Perturbed systems that are determined to be the most realistic, within whatever probability limits are established, will be used for the analyses that involve radionuclide transport and dose models.

The GSM is designed to be continuously refined and updated. Purposely generalized, the submodels are driven by input data that are in the form of probability density functions for data known to be stochastic or for data about which experts disagree on a value; the input data are in scalar quantities for data that can be quantified as single values; and the input data are in polynomial functions for variables whose values are dependent on other variables. These input data can be altered interactively by the user at the start of each simulation. Simulation models are site specific, and, although the submodels may have limited general applicability, the input data requirements necessitate detailed characterization of each site before application.

Introduction

Assessment of Effectiveness of Geologic Isolation Systems (AEGIS) is a research program established at the Pacific Northwest Laboratory (PNL) by the U. S. Department of Energy (DOE) to evaluate the performance of permanent nuclear waste repositories located in bedrock deep underground. The AEGIS program uses a coordinated set of relatively complex models (Fig. 1) to evaluate geologic repository performance at a specific location in a specific rock type over the length of time necessary for isolation of

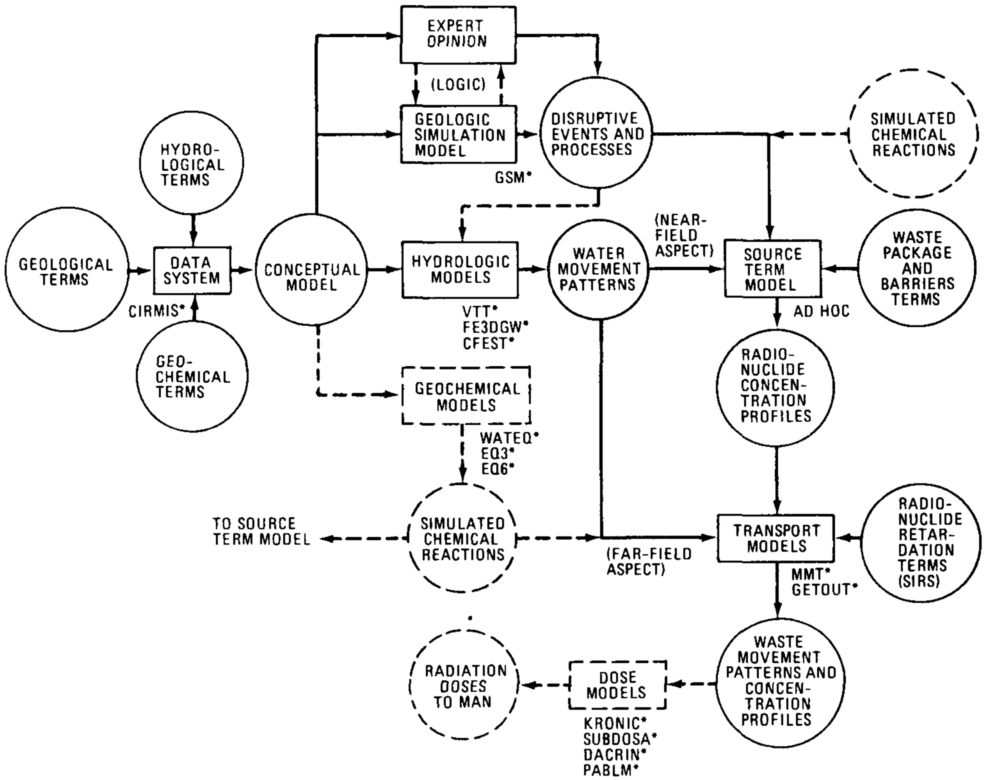


Figure 1 Schematic diagram of AEGIS analyses. Specific PNL computer codes (*) used in different aspects of the analyses are CIRMIS (data handling); GSM (geologic simulation model); VTT, FE3DGW, and CFEST (groundwater); MMT and GETOUT (radionuclide transport); WATEQ, EQ3, and EQ6 (geochemistry); and KRONIC, SUBDOSA, DACRIN, and PABLM (radiation dose to man).

nuclear waste. This paper discusses a quasi-deterministic, process-response geologic analysis model referred to as the Geologic Simulation Model (GSM). The GSM was developed by the AEGIS staff to assist in performance assessments by estimating future developments of the geologic/ hydrologic system. Application of the GSM to a hypothetical repository in basalts of the Columbia Plateau in eastern Washington has recently been completed. Results indicate that the GSM, when used properly with professional judgment, provides valuable understanding of the probable evolution of a geologic repository system.

The GSM is an integrated system of relatively simple models that ultimately provides perturbed system states as starting points for detailed analyses by the more complex hydrologic, transport, and dose models. To reduce uncertainties in the results obtained when the geologic processes are modeled, the GSM is site-specific. Figure 2 shows data available in a site-specific study for use in simulation modeling. In addition to providing general knowledge of geologic processes that may be acting, geologic

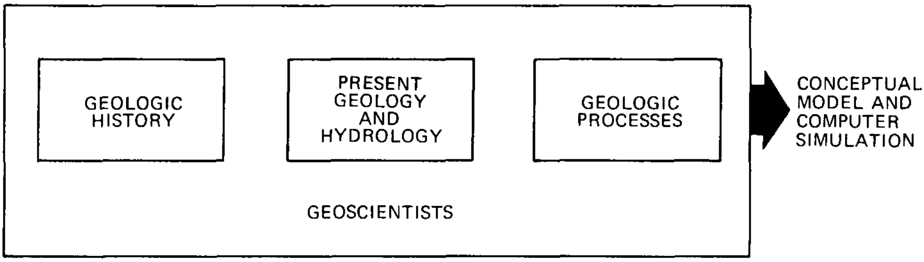


Figure 2 Conceptual components of the GSM.

mapping allows determination of current geology and hydrology as well as an estimation of geologic history. This allows scientists to determine the geologic processes actually operating and to estimate their rates. Conceptual modeling and eventual computer simulation can then be concentrated on the smaller subset of all possible geologic processes that act at a given site.

Table 1 shows the steps in developing a GSM for a specific area. Initial data collection includes geologic mapping, geophysical work, and determination of the hydrology and geochemistry of an area. A conceptual model of the area is constructed from these data and serves as the simplified basis for encoding the actual computer model. Data specific to the model needs are calculated or gathered during encoding of the computer model. These data will not necessarily be used in conceptualization, but they may be derived in part from the earlier data.

Debugging of the GSM and generation of preliminary results is a necessary step in model development. Results of early simulations are compared with the geologic history of the area modeled; experts review the results in the different aspects of geology and hydrology to suggest ways that the model should be altered so that it more accurately simulates reality. This calibration process makes the GSM able to simulate the past (as well as the future) so that model results can be compared with geologic history.

Following this calibration process, the GSM is operated in a Monte Carlo mode for several hundred simulations to generate possible future evolutions of the area modeled. Inputs to these Monte Carlo simulations are chosen to represent uncertainties in input data, and the number of Monte Carlo simulations performed is selected so that the simulations are extensive enough to explore uncertainties in the output parameters.

This manner of assessment does not predict the future; rather, it quantifies the geologic and hydrologic processes acting in an area as well as the local geologic and climatic history so that future geologic and hydrologic developments can be simulated. These simulations are not unique because many of the processes are stochastic at the current level of understanding. However, by outlining the range of possible future states of the system and the associated probabilities, we place quantifiable limits on the effects of geologic processes. This allows a smaller number of transport and dose analyses to effectively characterize the likely future behavior of the radionuclides in the repository.

Structure of the Model

The GSM uses a relatively low-resolution analysis of the far-field

Table 1 Steps Necessary in Constructing a GSM

	Hanford	Paradox Basin	Nevada T~est Site (Nevada Nuclear Waste Storage Investigations)
Initial data collection	Completed	Completed	Completed
Conceptualization	PNL-2892* (January 1981)	Discontinued	Completed
Model encoding	PNL-3542† (May 1981)		Some submodels completed
Model data collection	PNL-3542-2‡ (June 1982)		Completed
Model debugging	Completed		Completed
Results (preliminary)	Continuing		Completed
Expert review, revision	Continuing		
Final analysis	September 1981		

Stottlemyre, J. A., G. M. Petrie, G. L. Benson, and J. T. Zellmer, 1981, Assessment of Effectiveness of Geologic Isolation Systems. A Conceptual Simulation Model for Release Scenario Analysis of a Hypothetical Site in Columbia Plateau Basalts, *DOE Report PNL-2892, Battelle Pacific Northwest Laboratory, NTIS.*

†*Petrie, G. M., J. T. Zellmer, J. W. Lindberg, and M. G. Foley, 1981,* Geologic Simulation Model for a Hypothetical Site in the Columbia Plateau, *DOE Report PNL-3542, Pacific Northwest Laboratory, NTIS.*

‡*Foley, M. G., G. M. Petrie, A. J. Baldwin, and R. G. Craig, 1982,* Geologic Simulation Model for a Hypothetical Site in the Columbia Plateau: Volume II. Results, *DOE Report PNL-3542-2, Pacific Northwest Laboratory, NTIS.*

geologic and hydrologic systems and, currently, does not consider near-field effects by wastes in the repository; however, consideration of these effects may be added in the future. The GSM uses separate submodels, each of which addresses a specific type of geologic process or other event (Fig. 3). Each submodel separates the processes into categories for study by experts in different fields; this permits the experts to direct their attention in their areas of expertise without being distracted by other details of the entire geologic and hydrologic system. This division, combined with stepwise integration, also allows consideration of synergistic effects among different processes.

Consider the initial system state (Fig. 3) with the entire suite of processes acting upon it. If each process is addressed as a simple rate equation or, in the event that the physics of the situation is better understood, as a more complex function, the effect of each process on evolution of the system can be approximated by integration over a relatively short period of time. Summation of all the changes to the system gives a new system state as the initial condition for another such integration. If the time steps are kept short enough, nonlinear interactions among different processes can be closely approximated. Thus, even if interactions between processes are not

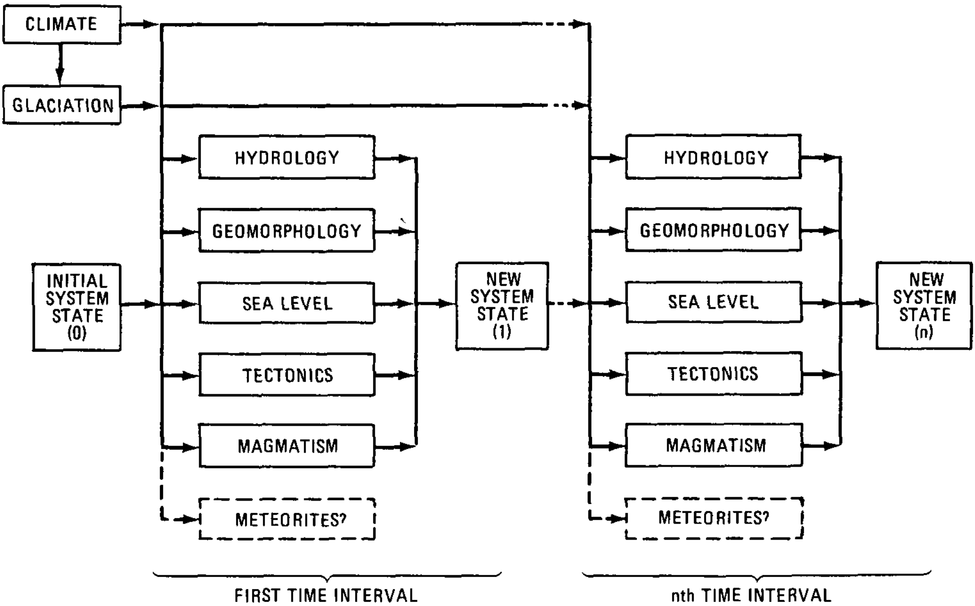


Figure 3 Schematic diagram of operation of the GSM.

well understood, the system model can be made to behave realistically. Such a result is not possible with modeling based on joint probabilities, because most joint probabilities are not known or must be considered as independent. Consequently modeling with joint probabilities does not allow for synergistic interactions.

Figure 4 shows the time steps used in the GSM that was constructed for million-year simulations of the Columbia Plateau. Hundred-year time steps are used for the first 20,000 years, which covers the 10,000-year period currently receiving institutional attention and a time period for which radiocarbon data of geologic history are the most abundant. The next 180,000 years are covered by 1000-year time steps. Knowledge of the past does not warrant the detail for this 180,000 years that was employed for the first 20,000 years modeled; however, our knowledge of the past is sufficient to maintain better resolution than that used for periods beyond

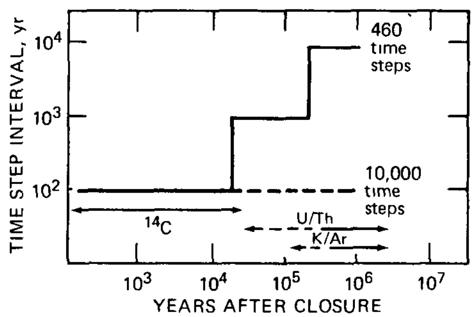


Figure 4 Progressively increasing time steps used in the GSM.

200,000 years. The interval from 200,000 to 1,000,000 years is modeled using 10,000-year time steps. We do not think that the GSM is especially prophetic this far into the future. However, our experience has been that simulations of geologic processes that are not carried to the order of one million years into the future may have inherent inaccuracies or instabilities that do not appear in the first few thousand years of simulation, but are apparent at the longer times. The million-year simulation time is long enough to have these errors become apparent and to accommodate any institutional requirements that may develop; however, it is short enough to avoid unpredictable (with current technology) changes in the lithospheric plate configuration or rates of plate movements.

An exception to this progressively increasing time step sequence is made for modeling of climate and continental glaciation (Fig. 3). The simulated climate, as is the case with the actual Late Tertiary climate, is relatively unstable. Thus simulation of continental glaciations, which can be calibrated against past glaciations, is unstable enough without trying to accommodate changing time step lengths. Because no feedback to the climate or glaciation submodels occurs from the other submodels, running these two submodels separately with homogeneous hundred-year time steps is reasonably economical. These results are then fed into simulations with the other submodels. This still saves computer time because the majority of the submodels are run with increasing time intervals. Running the submodels in increasing time intervals results in 460 time steps; whereas 10,000 time steps are necessary for homogeneous hundred-year time intervals.

The GSM is designed to be driven by input data. Individual submodels are as deterministic as is practical within the constraints of current knowledge of geologic processes and the necessities for providing data. Figure 5 shows the relation between input data, the model, and the output packages. Input data in the form of polynomials, scalars, probability density functions, raw climatic data, and layer data that describe the geologic framework are controlled by the input package. This package employs interactive computer graphics to allow an investigator unfamiliar with computers to change input data and observe the effects on model output.

Separation of the geologic system submodels and the input data allows a balance between the deterministic submodels and the more stochastic input data that quantify uncertainties in the overall system model. This separation facilitates calibration of the submodels by allowing simple, interactive changes of input data rather than requiring changes to the submodel codes. The result is a quasi-deterministic model that incorporates as much physics as is practical and accommodates uncertainties in understanding of the processes by employing probability density function inputs or probability density function uncertainty terms in scalar and polynomial data.

The output packages manage the overwhelming quantity of data generated by the model (46,000 numbers for 100 output variables for 460 time steps in each simulation). A million-year simulation takes two to three minutes on a mini-computer; therefore a Monte Carlo analysis consisting of one thousand simulations can generate 46 million data. A statistical package summarizes these data in the form of probability density functions for output variables, correlation coefficient matrices, and other tools (described in the "Output" section that appears later in this paper).

Climate and Glaciation

The GSM developed for a hypothetical site in the Columbia Plateau is described by Petrie et al. (1981) and Foley et al. (1982). Discussion of a few representative submodels will help illustrate how the GSM simulates geologic processes. Our experience has been that the climate and glaciation submodels have a major effect on

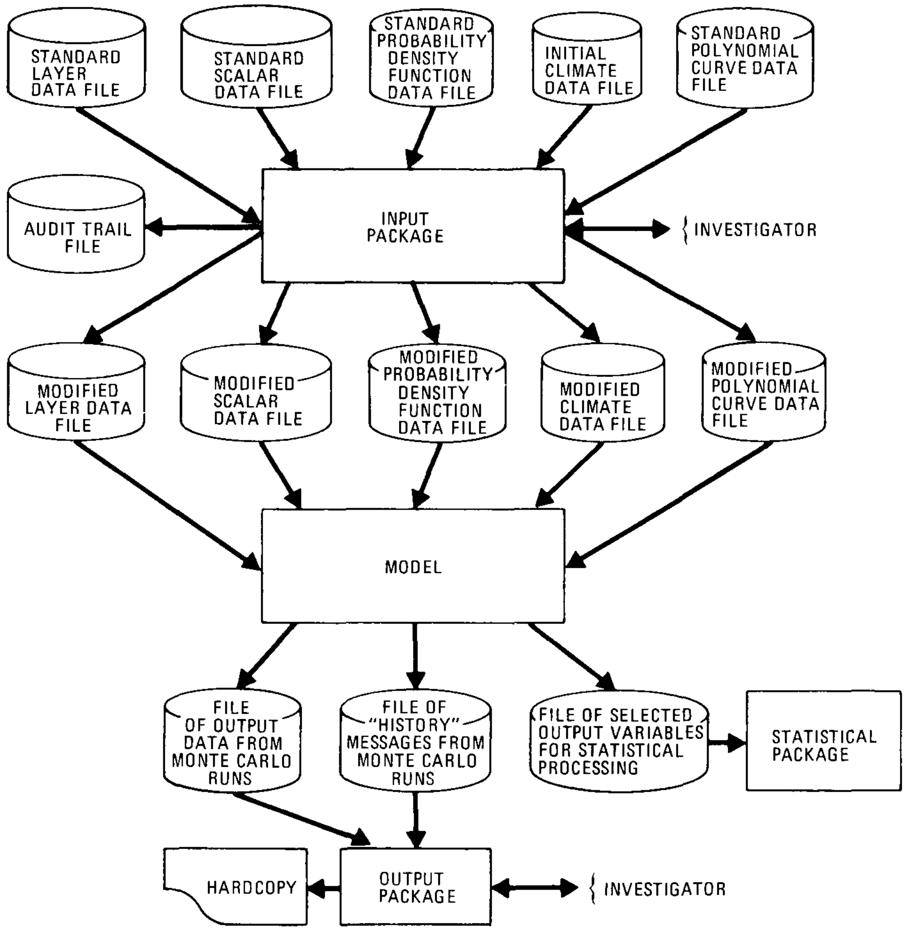


Figure 5 Schematic diagram of the three main parts of the GSM—input, model, and output.

GSM analyses. Therefore these submodels will be discussed in detail, and a briefer discussion of the hydrology submodel will follow.

The basis for the climate submodel is Milankovitch's hypothesis (Kukla et al., 1981). Figure 6 shows the three basic parameters of this hypothesis:

1. Variation of the orbital tilt, E, of the earth relative to the plane of the ecliptic, E
2. Precession of the orbital tilt axis, T, relative to the major axis of the earth's orbit
3. Eccentricity, e, of the earth's orbit.

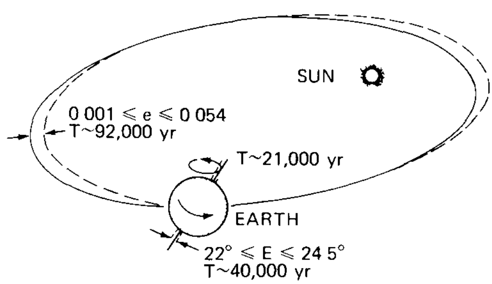


Figure 6 Milankovitch driver for the climate submodel.

As shown in Fig. 6, these parameters have variational periods, and, according to the hypothesis, these varia-

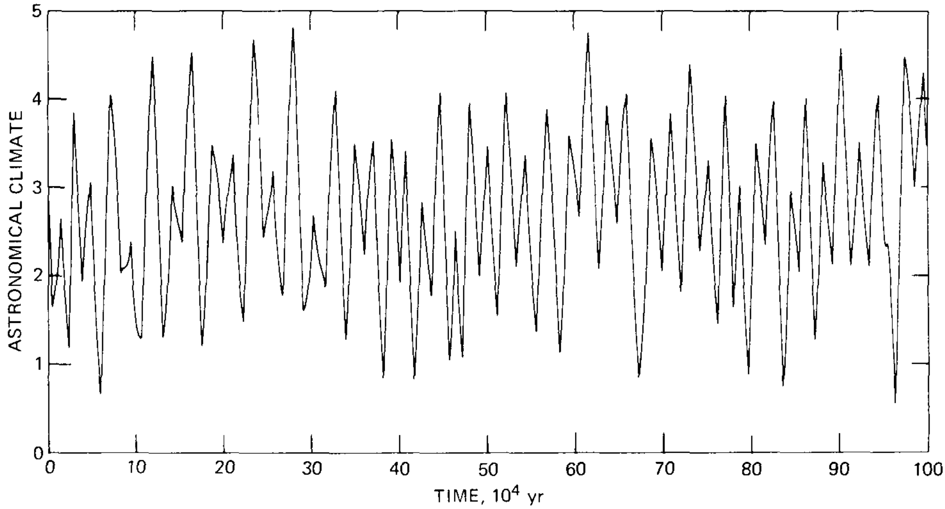


Figure 7 Future astronomical climatic index (ACLIN).

tional periods result in a quasi-periodicity of 100,000 years in the global climate. Shorter-period fluctuations are superimposed on this 100,000-year periodicity (Fig. 7), and the resulting astronomical climatic index (ACLIN) may be a relatively accurate predictor of colder climates (low ACLIN number) and warmer climates (high ACLIN number). Advantages of ACLIN are that it can be correlated and calibrated against past climates (Fig. 8), and, because it is based on predictable orbital parameters, ACLIN gives a basis for predicting future climates. Results of ACLIN are shown in Fig. 9, which is a quantized climate distribution varying from a climate index number of 1 (interglacial) to 5 (coldest stadial). Intermediate states are climate index numbers of 2 (temperate interstadial), 3 (interstadial), and 4 (stadial). Superimposed on the climate distribution is a random volcanic ash perturbation that is a probability-density-function estimator based on data (Fig. 10) on past volcanic eruptions. Effects of the volcanic ash perturbation are most apparent in the GSM in a climate

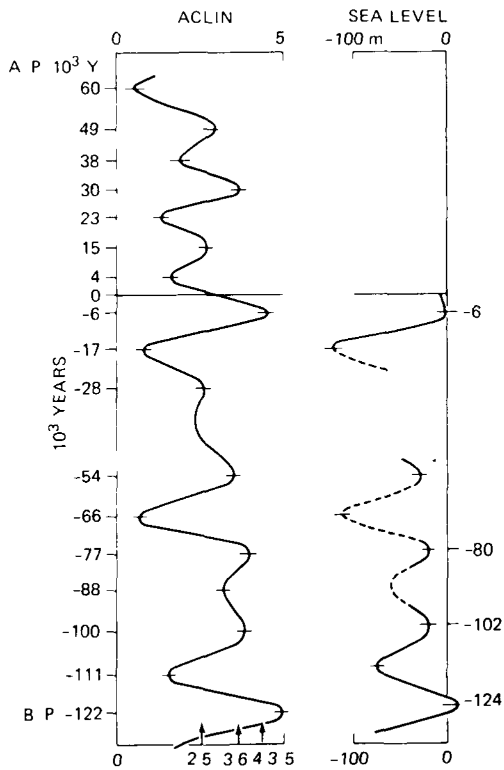
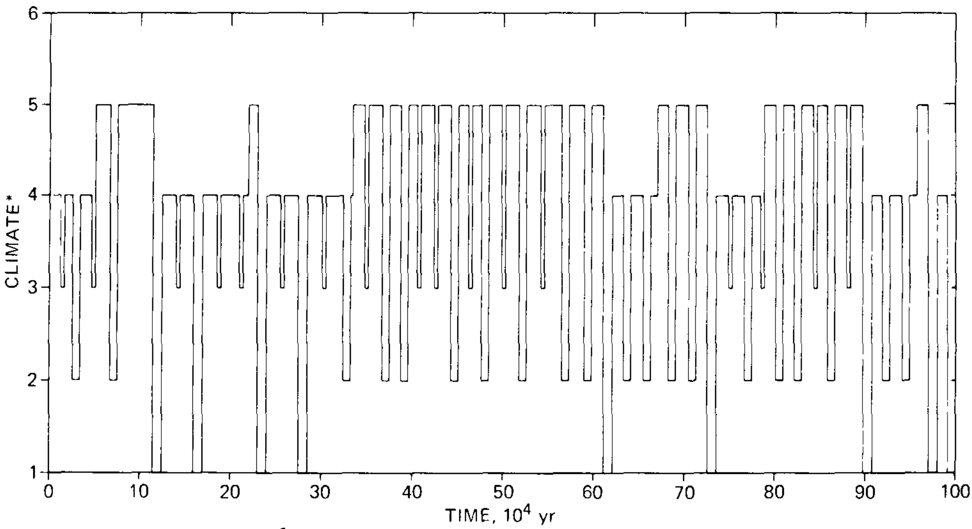


Figure 8 Calibration of ACLIN against Pleistocene sea-level data.



*Warmest is represented by 1; 5 represents the coldest.

Figure 9 Quantized climate states used in the GSM.

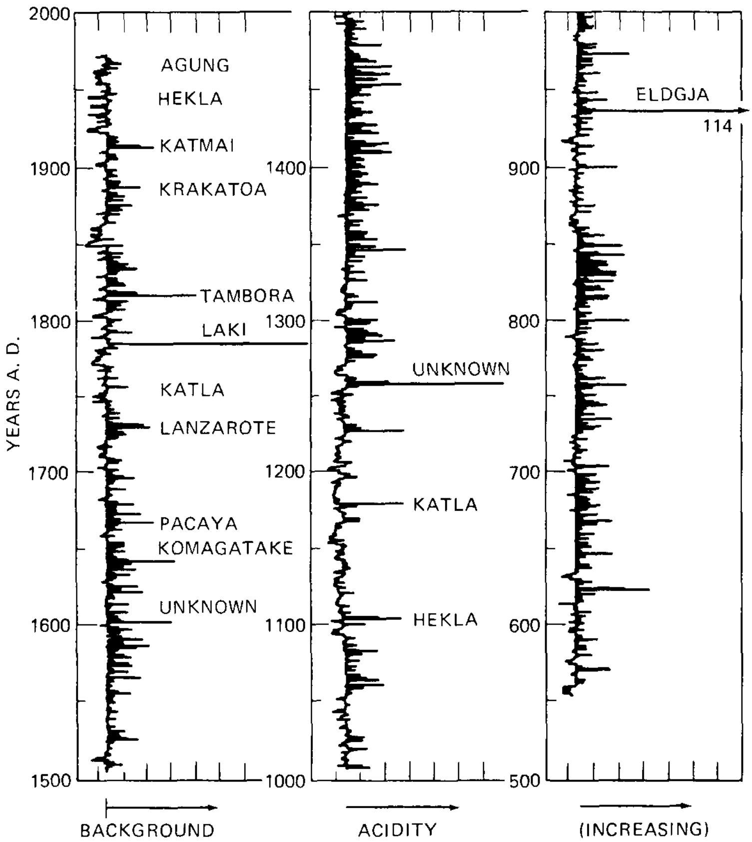


Figure 10 Past volcanic activity used to generate a probability density function for stochastic climate perturbations.

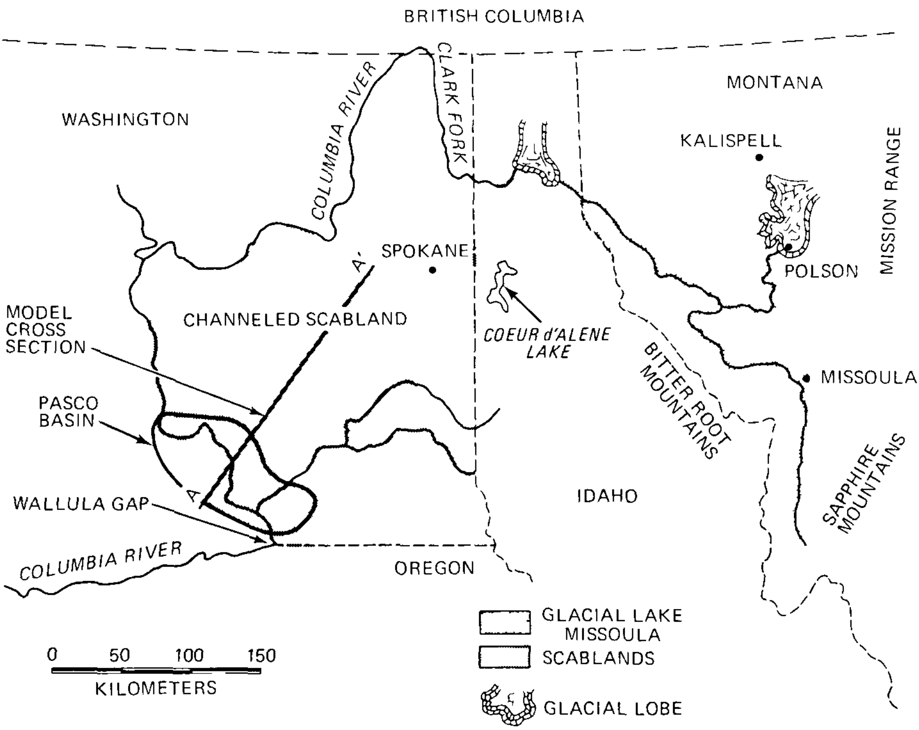


Figure 11 Index map of Columbia Plateau showing Pleistocene glacio-fluvial features.

state of 3. In this state, the effects of a major eruption can change the climate to full glacial.

The climate index may be used to estimate future rainfall and recharge relative to those at present. However, on the Columbia Plateau, the climate state also must be used in a more complex fashion to simulate the behavior of continental glaciers because these glaciers have a profound effect on the hydrology of the area selected for study as a potential nuclear waste repository. Figure 11 shows the Pleistocene features of the Columbia Plateau and the location of the Pasco Basin, the area of major interest. In the past, several lobes of the Cordilleran ice sheet have advanced southward to within a few hundred kilometers of the Pasco Basin; and, in the future, these lobes could do so again. Figure 11 shows one such lobe that blocked the Clark Fork of the Columbia River and resulted in ponding of a large glacial lake (Lake Missoula) in eastern Idaho and western Montana. This ice dam failed repeatedly and resulted in catastrophic flooding across the eastern Columbia Plateau and into the Pasco Basin. It is reasonable to expect that in the future the hydrologic system of the eastern Columbia Plateau probably will be affected by glaciers, glacial outwash and associated sediments, and catastrophic flooding of the Lake Missoula type.

The GSM must estimate such future behavior as accurately as possible. A way to ensure this is to calibrate the glaciation submodel against the Pleistocene by comparing the behavior of the submodel based on Pleistocene ACLIN data with the reconstructed behavior of Pleistocene

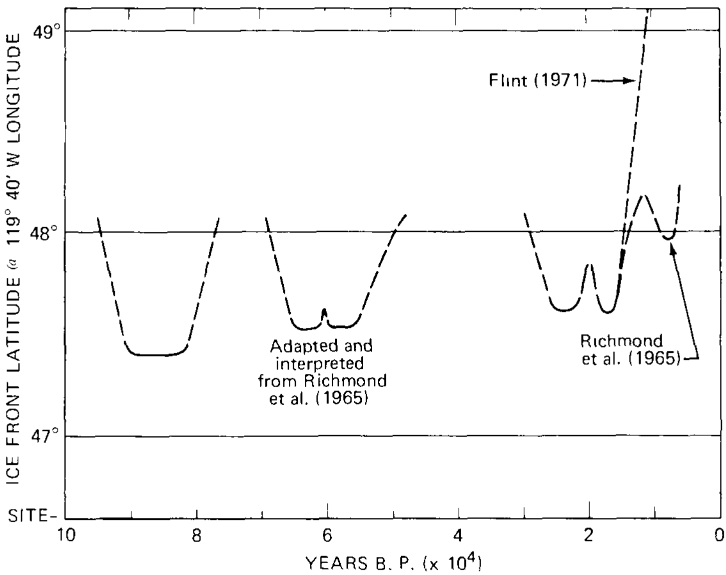


Figure 12 Late-Pleistocene advances of the Okanogan lobe of the Cordilleran ice sheet.

Table 2 Basic Equation of the Glaciation (Okanogan Lobe) Submodel

Basic Equation: Glacial rate = Δ Climate $\times$ Nominal rate $\times$ e^{-D}

where

Glacial rate =	Rate of glacial movement (km/100 yr)
Δ Climate =	Change in climate state (dimensionless)
Nominal rate =	100 to 200 m/yr
e =	Base of Naperian logarithms (dimensionless)

$$D = C_1 \left(\frac{R}{R_{QMax}} \right)^{C_2} + C_3 \Delta \mathrm{Time}$$

$C_1 \sim$	2×10^{-3} (constant coefficient, dimensionless)
$C_2 \sim$	70 (constant coefficient, dimensionless)
$C_3 \sim$	2×10^{-1} (constant coefficient, 1/yr)
R =	Radius of the ice sheet (km)
R_{QMax} =	Maximum quaternary radius of the ice sheet (km)

glaciers. Figure 12 shows the probable positions of end moraines of the Okanogan lobe of the Cordilleran ice sheet during the past 100,000 years. Early attempts using simple climate-state-linked glacial rate equations for the Okanogan lobe allowed the GSM to simulate Pleistocene behavior, but glacial activity predicted for the future was far more severe than that of the Pleistocene. As a result, the equations shown in Table 2 were

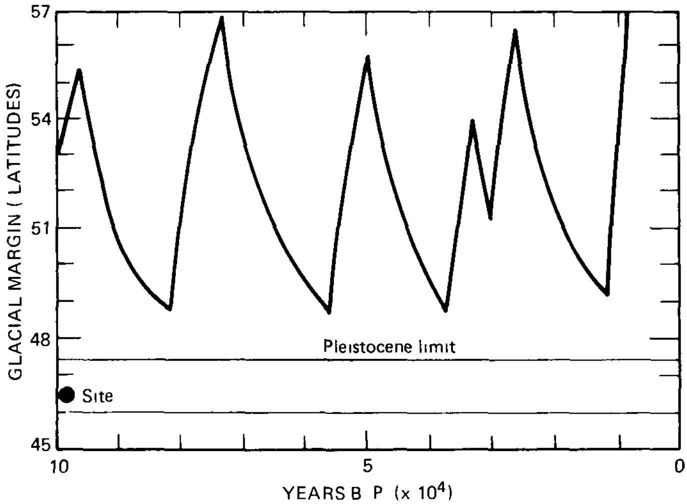


Figure 13 Calibrated Pleistocene behavior of the glaciation submodel.

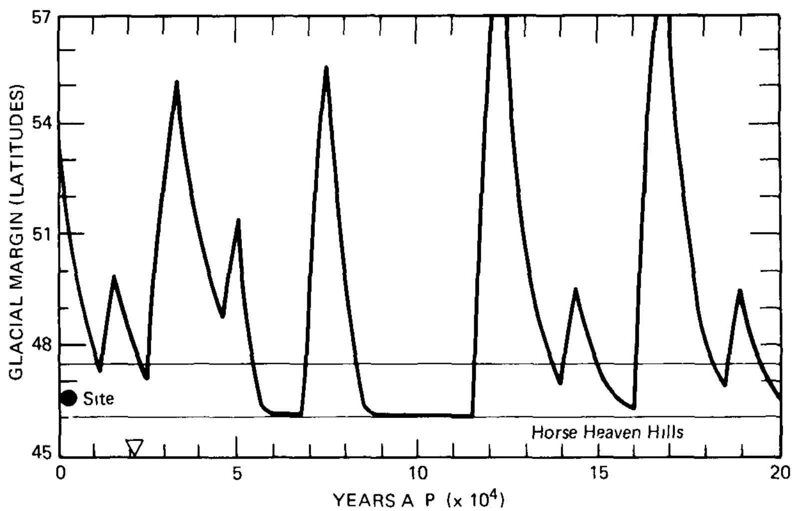


Figure 14 Future behavior of the glaciation submodel.

developed, and a much more sophisticated glacial rate model than was originally thought necessary was developed. This model, when calibrated, can simulate the Pleistocene relatively well (Fig. 13); however, it does no better than the less sophisticated rate equations used earlier. Nevertheless, the more sophisticated model was necessary to demonstrate that the effect was not an artifact of simplistic simulation. All the major advances of Fig. 12 are shown in Fig. 13 along with an extra that is inherent in ACLIN. Figure 14 shows a part of the future behavior of the Cordilleran ice sheet based on the calibrated continental glaciation submodel. By 55,000 years into the future, the Okanogan lobe apparently will overrun the hypothetical repository site in the Pasco Basin. This result is dependent primarily upon ACLIN and not upon the way that the Cordilleran ice sheet is simulated as long as the simulation is calibrated against the Pleistocene.

The interactive submodel development and analysis procedure, which resulted in more sophisticated attempts at a glacial submodel as results were compared with Pleistocene behavior, is an example of how each submodel is calibrated. A further geographical constraint on the glaciers was applied in the glaciation submodel. The Horse Heaven Hills, an anticlinal ridge forming the southern boundary of the Pasco Basin, was considered to have the potential for halting advance of a Cordilleran ice sheet. Thus a decay term that effectively halted the glaciers at the Horse Heaven Hills was introduced into the rate equation. This is a site-specific rate equation because topographic effects are included in a relatively simple fashion compared to that necessary to incorporate them deterministically.

This result was not anticipated, and it is an example of the use of the GSM for focusing study of the future behavior of an area chosen for a nuclear waste repository. Expert consultants considered the possibility that future glaciations would be more severe than past ones. However, estimates of the probability of future glaciers overrunning the hypothetical repository were almost entirely subjective. The estimate by the GSM is much less subjective, and it indicates that the probability of future severe glaciations resulting in ice overlying the Pasco Basin is nearly 100%. Further, advance of future glaciers to the Horse Heaven Hills probably will result in extrusion of rapid valley glaciers through Wallula Gap (Fig. 11) (similar to the valley glaciers that drain the eastern margin of the Greenland Ice Cap). Wallula Gap probably will be enlarged substantially even if, contrary to the assumption in the GSM, the ice sheets overrun the Horse Heaven Hills. Wallula Gap is the only surface water outlet of the Pasco Basin, and, in the past, it has acted as a hydrodynamic choke on discharge of Missoula flood flows from the Pasco Basin. Consequently net deposition of glacio-fluvial sediments has resulted in the Pasco Basin during Missoula floods; this compensates for any catastrophic flood erosion of the basalts and protects a hypothetical repository from further fluvial erosion. However, if a glacier is extruded through Wallula Gap, enlargement of the gap may prevent similar hydraulic damming in the future, and the Pasco Basin may become an area of net erosion rather than its present state of net deposition.

Hydrology

The basis for performance assessment of a hypothetical repository is the performance of the hydrologic system. Except for volcanic activity and meteorite impact, movement of radionuclides out of the repository and into an area where they can be hazardous is in groundwater flow. The GSM employs a relatively simple hydrologic model because of all the other phenomena that must be modeled. Table 3 shows the basic

Table 3 Basic Equation of Hydrology Submodel

Basic Equation

$$v = K\nabla H \text{ (Darcy's law)}$$

v = Darcy velocity (specific discharge)
K = Hydraulic conductivity
H = Head

Table 4 Factors Affecting Hydraulic Conductivity

Faulting (brecciation)
Shaking (seismic, meteorite impact)
Folding
Glacially induced fracturing
Secondary mineralization
Periglacial processes

Table 5 Factors Affecting Potentiometric Head

Climate—recharge
River entrenchment
Uplift
Erosion
Glacial blockage

equation, Darcy's law, and the two basic parameters necessary to drive it: hydraulic conductivity and the potentiometric head. Table 4 shows the geologic processes that affect hydraulic conductivity. These processes are modeled quite simply in the GSM; however, in many cases, the models in the GSM must be supported by more detailed models. Therefore the GSM is a simple, abstracted version of more complex considerations of the effects of, for example, faulting and seismic shaking on hydraulic conductivity where the basis for simplification is geophysics and fracture flow analysis.

Table 5 shows the processes that affect the potentiometric heads driving the hydraulic system. For a given site such as the Columbia Plateau (Fig. 15), investigators use a complex hydrologic model to deduce the groundwater flow patterns that may affect a potential repository. In the case of the Columbia Plateau, the complex modeling allowed selection of a model cross section (Fig. 15, AA′) that paralleled the deep, confined flow, and that, therefore, could be assumed to have nearly two-dimensional flow. Figure 16 shows the model cross section used in the GSM for the Columbia Plateau. Also shown are the general regional flow paths and the way that the relatively complex regional basalt stratigraphy was simplified into two major confined aquifers that have a separating confining zone and are overlain by an unconfined aquifer in the Pasco Basin. Further simplification is made possible by considering flow along the flow lines as a one-dimensional model that effectively simulates the three-dimensional flow system of the Columbia Plateau as it affects the Pasco Basin.

Because a porous flow model was being used to describe a fracture flow system, selection of input data for the hydrology submodel was difficult. It was necessary to derive hydrologic parameter data from well measurements, which are macroscopic data acquired from a megascopic flow system. Figure 17 shows a probability density function derived from hydraulic conductivity data of basalts in the major recharge area of the repository horizon. Simultaneous with the derivation of the hydrologic parameter data, investigators were calibrating the more complex flow model against available groundwater data; this resulted in a calibrated hydraulic conductivity four orders of magnitude smaller than that from well data. This exercise emphasizes the need to have GSM development simultaneous with the application and calibration of more complex models as well as the need to check and calibrate the GSM submodels against those more complex models in each step of the simulation.

Output

As discussed earlier, the GSM has a potential for producing an overwhelming amount of data. The statistical package is used as a primary data summarization tool to

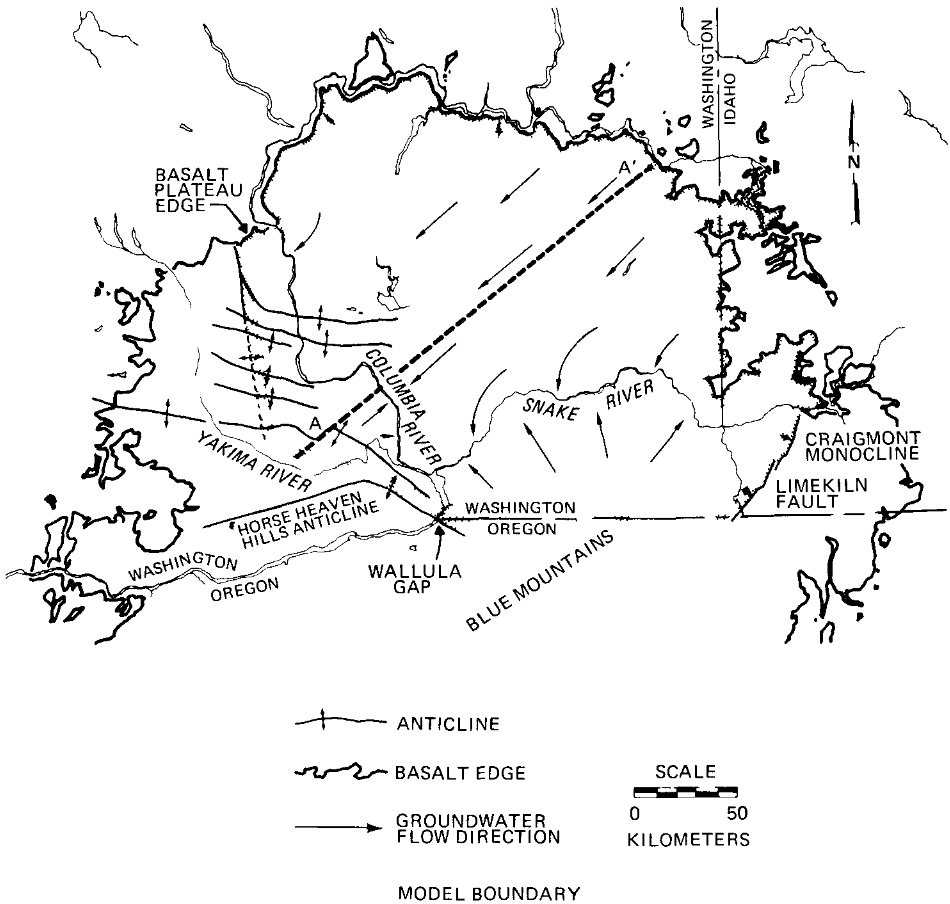


Figure 15 Index map of hydrologic features of Columbia Plateau.

display each variable as a probability density function and cumulative probability distribution (Fig. 18). This summary is an effective way of describing the behavior of each variable in the entire Monte Carlo simulation; it also relates values of the variables to any institutional requirements for probabilities not exceeding a certain value. However, such summaries do not allow evaluation of behavior of the submodel, and such evaluation is a necessary step in determining if the output data are reasonable.

Figure 19 and Table 6 illustrate other ways in which the output data can be displayed to facilitate evaluation of the results. Figure 19 is a scatter diagram of two variables that may be related. Table 6 shows two 5×5 matrices of correlation coefficients between pairs of variables. These two tools allow comparison of apparent relations between variables to the conceptual model encoded in the GSM submodels. Analyses of these relations can lead to quantification of unsuspected interactions between variables. However, as Table 6 shows, some of the apparent relations may be spurious. This is because two variables with monotonic behavior as

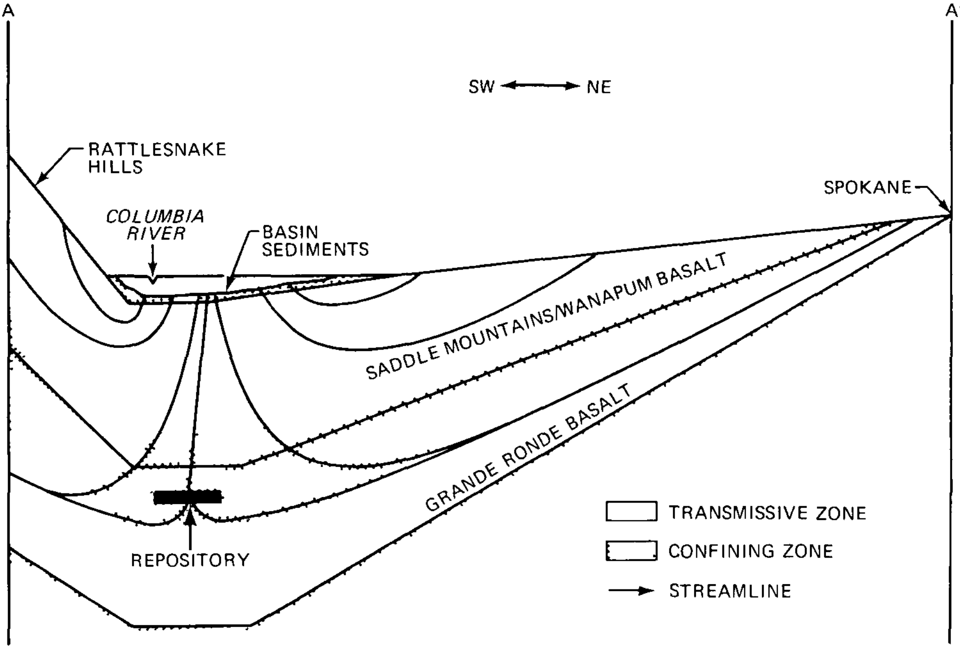


Figure 16 Model cross section used in the GSM.

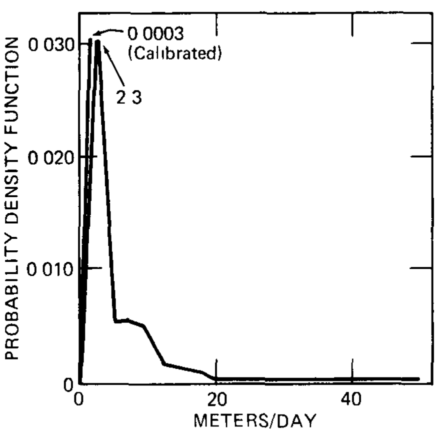


Figure 17 Hydraulic conductivity of Grande Ronde basalts in the northeast recharge area of the Columbia Plateau GSM.

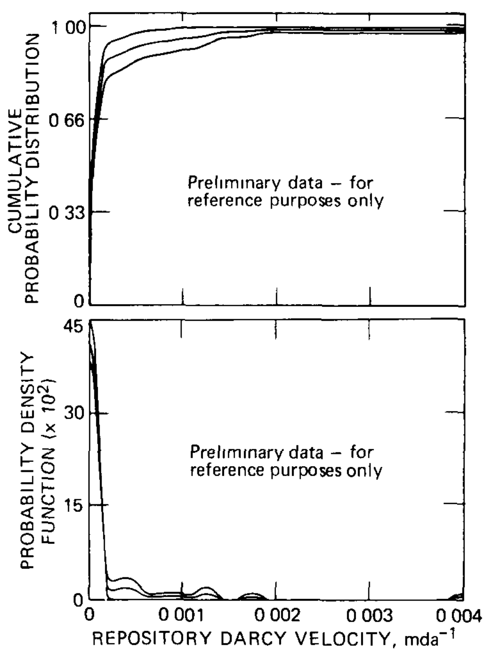


Figure 18 Probability density function and cumulative probability distribution—representative GSM output.

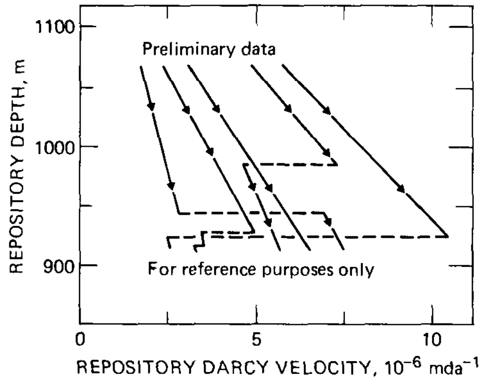


Figure 19 Scatter diagram— representative GSM output.

Table 6 Correlation Coefficient Matrices—Representative GSM Output

	RDEPTH*	CHRS 1*	DVRPOS*	FLHCSW*	SUMFTN*
RDEPTH	†	−0.3†	−0.8	−0.4†	−0.06†
CHRS 1		†	+0.3	+0.2†	+0.05†
DVRPOS			†	+0.3†	+0.06
FLHCSW		Symmetric		†	+0.05†
SUMFTN					†
	RDEPTH	**RKMSIT***	**DVRPOS**	**SLOPE***	**ICLIM***
RDEPTH	†	+0.6†	−0.8	−0.5	+0.2
RKMSIT		†	−0.5	−0.9	+0.3
DVRPOS			†	+0.4	−0.2
SLOPE		Symmetric		†	−0.3
ICLIM					†

**Variables: RDEPTH = Depth to repository*
CHRS 1 = Change of repository hydraulic conductivity caused by basement faulting
DVRPOS = Darcy velocity through repository
FLHCSW = Change of southwest recharge area hydraulic conductivity caused by faulting in SW area
SUMFTN = Number of local faulting events near repository
RKMSIT = Distance from repository to edge of Cordilleran ice sheet
SLOPE = Slope of Columbia River
ICLIM = Climate state (stadial, interstadial, temperate interstadial, interglacial)

†Spurious correlation, usually caused by monotonic behavior of both variables as functions of time.

functions of time may appear to be strongly correlated when in fact they are not physically related at all.

Thus the statistical package for GSM output data is in a period of further development. More sophisticated tools are being developed to filter out the dominating time-series behavior of many of the correlations between variables and to better detect more subtle and meaningful interactions between variables.

Conclusion

The AEGIS Geologic Simulation Model, in its application to a hypothetical repository in basalts of the Columbia Plateau, has demonstrated its usefulness for repository performance assessments. In constructing a GSM for a particular location, subjective and objective considerations can be separated, subjective system evaluations can be reduced, and perceptions of the way the geologic/hydrologic system should operate can be quantified. Operation and calibration of the GSM, with feedback from geologists, isolates critical parameters that apparently will dominate future system behavior. This, in turn, helps indicate particular areas where further work is necessary. Thus the GSM does not predict the future, but its simulations of possible futures focus attention and detailed analyses on the controlling processes.

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Environmental Issues of Repository Licensing: An Evaluation of a Hypothetical High-Level Radioactive Waste Repository

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This paper presents results of an environmental assessment conducted under the direction of the Office of Nuclear Waste Isolation as part of the National Waste Terminal Storage program. The study defined a range of potential environmental effects of constructing, operating, decommissioning, and long-term isolation of a nuclear waste repository.

The analytical methodology used to determine potential environmental effects required definition of a hypothetical environmental setting and repository. Potentially applicable regulatory requirements were identified and were used as guidelines to evaluate permitting feasibility. The environmental effects of repository development were analyzed for the two major time periods of concern: short term (the period of construction, operation, and decommissioning) and long term (the isolation period after decommissioning). As a result of this analysis, major environmental uncertainties and issues were identified.

Introduction

Identifying environmental issues related to high-level nuclear waste storage is an important part of developing a program to construct and operate a nuclear waste repository in a deep geologic formation. Many federal, state, and local agencies will be involved in the licensing and permitting process. Early identification of potential issues is valuable in providing programmatic direction for the efficient and timely use of resources in the licensing program. Such a strategy leads to early development of research studies addressing the issues and provides guidance in the interactive process of promulgating guidelines, standards, and criteria for the licensing of repositories.‡ In a study conducted to identify important issues, a hypothetical repository at a hypothetical site was evaluated within the framework of the requirements of a license application to the Nuclear Regulatory Commission.

This paper presents results of an environmental assessment conducted under the direction of the Office of Nuclear Waste Isolation (ONWI) as

‡The Nuclear Regulatory Commission and the Environmental Protection Agency are responsible for promulgating guidelines, standards, and criteria; however, these agencies seek counsel in developing the regulations. Draft regulations issued for comment, seminars and workshops, and other mechanisms are part of the current interactive process.

part of the National Waste Terminal Storage (NWTS) program. The study defined a range of potential environmental effects of constructing, operating, decommissioning, and long-term isolation of a nuclear waste repository in a deep geologic formation. The analysis identified what are currently perceived as important environmental issues to be addressed in preparing to license the repository.

The paper first presents a summary of the major issues identified by the analysis. This section is followed by a discussion of the analytical methodology used, including a brief description of the hypothetical environmental setting and of the repository. Potentially applicable regulatory requirements are summarized in the next section. Finally, environmental effects of the repository during the two major time periods of concern (short term and long term) are presented, and the major issues associated with each period are identified.

Conclusions

The analysis indicated that the short-term effects are similar to those of any large industrial project located in a rural area:

- The influx into a rural area of a relatively large labor force for constructing and operating the repository will result in major beneficial and adverse socioeconomic effects.
- There will be some adverse effects on wildlife and land use, including commercial uses (forestry and agriculture), as a result of the large surface land area committed to actual structures and facilities (over 162 ha, 400 acres).
- Effects of air emissions, solid waste, wastewater effluents, and noise will be within an acceptable range.

Long-term effects, more difficult to predict over the long time frame, include:

- The effects of heat and subsidence, under normal conditions, will be the primary impacts of waste isolation over the long term.
- Postulated disruptive phenomena (natural, human-induced, and/or waste-induced) are classified as "incredible" events, i.e., events with probabilities of less than 1 in 10 million.
- Some unlikely, though possible, phenomena, such as groundwater flow directly through the repository, could result in radiation doses that would be a fraction of those delivered by natural background radiation.

The environmental impact analysis identified the following issues as potentially important considerations in the licensing of a nuclear waste repository:

- Handling, storage, and disposal of mined rock are major concerns, particularly in the geologic medium of salt, which in large quantities is toxic to plants and animals.
- Potential effects of thermal releases on surrounding media (i.e.. thermal expansion and temperature rise at the surface) and on aquatic and terrestrial biota are presently not well defined. Therefore very conservative design limits are being used.
- Transportation of large quantities of high-level radioactive waste is a potential public concern.
- Potential methods (tax incentives, construction camps, etc.) must be developed to mitigate adverse socioeconomic impacts induced by the influx of the construction and operation work force and by the severe reduction in local employment when the facility is decommissioned.
- Methods must be developed to reduce the risk of potential human

interference with repository performance during the isolation phase (up to 10,000 yr).

- Public perception of radiation as a potential health and safety issue must be addressed.

Projects addressing many of these issues are now part of the ongoing NWTS program. As an example, alternative measures to mitigate the adverse socioeconomic effects of repository development are already being studied by the ONWI Systems Department. A Licensing Group task force is studying the issue of human interference and will prepare a series of topical reports that address public and regulatory agency concerns about human intrusion. Other issues, such as research into the effects of temperature increases on soils and biota and the effects of windblown salt, will be evaluated at a later date. Issues related to public concern over potential radiation exposures will be continuously addressed by means of public interaction and education, now part of the Department of Energy (DOE) consultation and public participation process.

Analytical Methodology

As part of the ONWI licensing program, an analysis was performed to provide an example approach to the documents that will be required for a license application to the Nuclear Regulatory Commission (NRC) for a nuclear waste repository. This analysis consists of two parts; the first is similar in scope to a safety analysis report, and the second is similar to an environmental report. The safety and environmental evaluations were conducted by systematically identifying and reviewing state-of-the-art repository information and incorporating that information into the format of the major documents required by the NRC for licensing nuclear facilities. The overall purposes of the evaluation were to provide ONWI with information on how well the technical program would meet licensing requirements, as such requirements were then perceived; to identify important licensing issues; and to provide DOE with a framework document to facilitate discussion of the issues and requirements.

The environmental evaluation presented the data needs and methodologies for evaluating environmental effects. The study analyzed potential effects resulting from repository construction, operation, decommissioning, and long-term isolation and also identified major environmental uncertainties and issues.

A hypothetical or "reference" environmental setting was defined to provide a realistic assessment of potential effects, and a conceptual or reference repository was described. Potential environmental effects were then identified for the two major phases of the project: the short-term phase, which includes construction, operation, and decommissioning of the repository, and the long-term phase, which is the isolation period after decommissioning and closure. Because of the lack of specific data for a real site, potential effects were not quantified, but a range of effects was bounded where possible, and unresolved environmental issues requiring further study were identified.

Reference Site. To assess potential effects, investigators defined an environmental setting. A repository in a salt dome somewhere in the Gulf interior region (Texas, Louisiana, or Mississippi) was chosen for this analysis. The reference site was a composite of surface and subsurface data gathered from published reports on industrial sites in the Gulf area. The data required for site characterization and impact assessment were

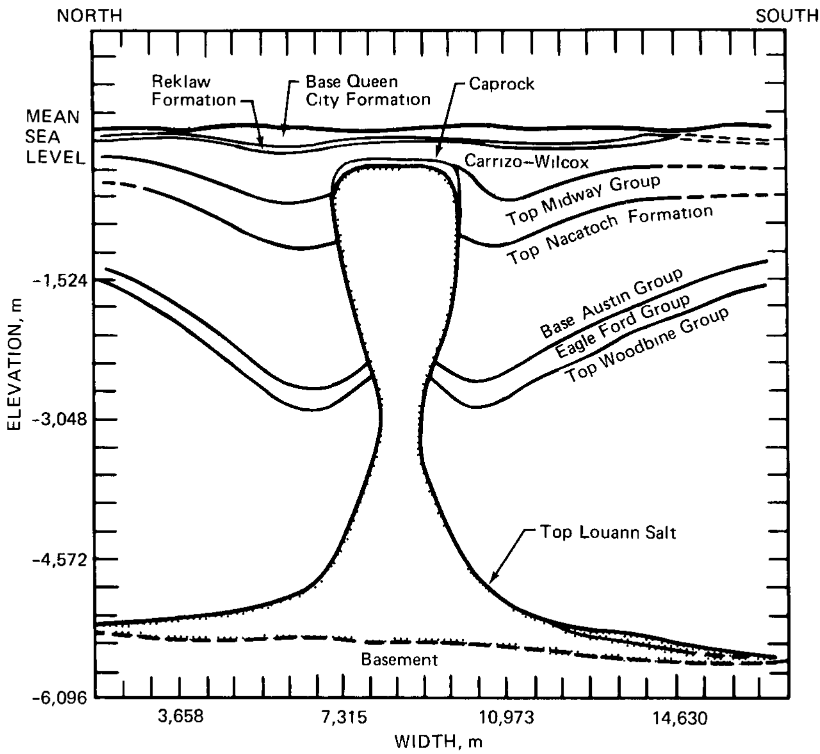

Figure 1 Cross section of reference dome.

assumed to be gathered from a baseline measurement program. The description of a representative or reference salt dome (Fig. 1) was derived from information on a dome in the eastern Texas salt dome basin. This area has been geologically stable for approximately 50 million yr (since the Reklaw formation). The geologic region around the reference dome, approximately 1400 km^2 (540 mi^2), was described as consisting of rolling hills. The regional surface slopes generally from northwest to southeast. The reference dome pierces 5000 m (16,000 ft) of strata, as shown in Fig. 1, ranging from the Late Jurassic to the Early Tertiary periods in geologic age. A cap rock with a thickness ranging from 15 m (50 ft) to more than 75 m (250 ft) covers the top of the dome. It is composed of a top zone of pyrite, a middle zone of gray shaley limestone, and a lower zone of clear, very dense anhydrite. The dome itself is made of crystalline halite, with some evidence of shale inclusions on its periphery. At the repository depth (640 m or 2100 ft), the dome has a cross-sectional area of approximately 850 ha (2100 acres).

The representative ecological setting (Fig. 2) included several vegetation types, e.g., mixed hardwood forests and cleared agricultural and pasture lands. Surface water bodies near the site included a river, a creek, and a swamp. One major aquifer provided water to the larger municipalities and industries in the region. Typical flora and fauna associated with each habitat were described, including one threatened and endangered species, the American alligator (*Alligator*

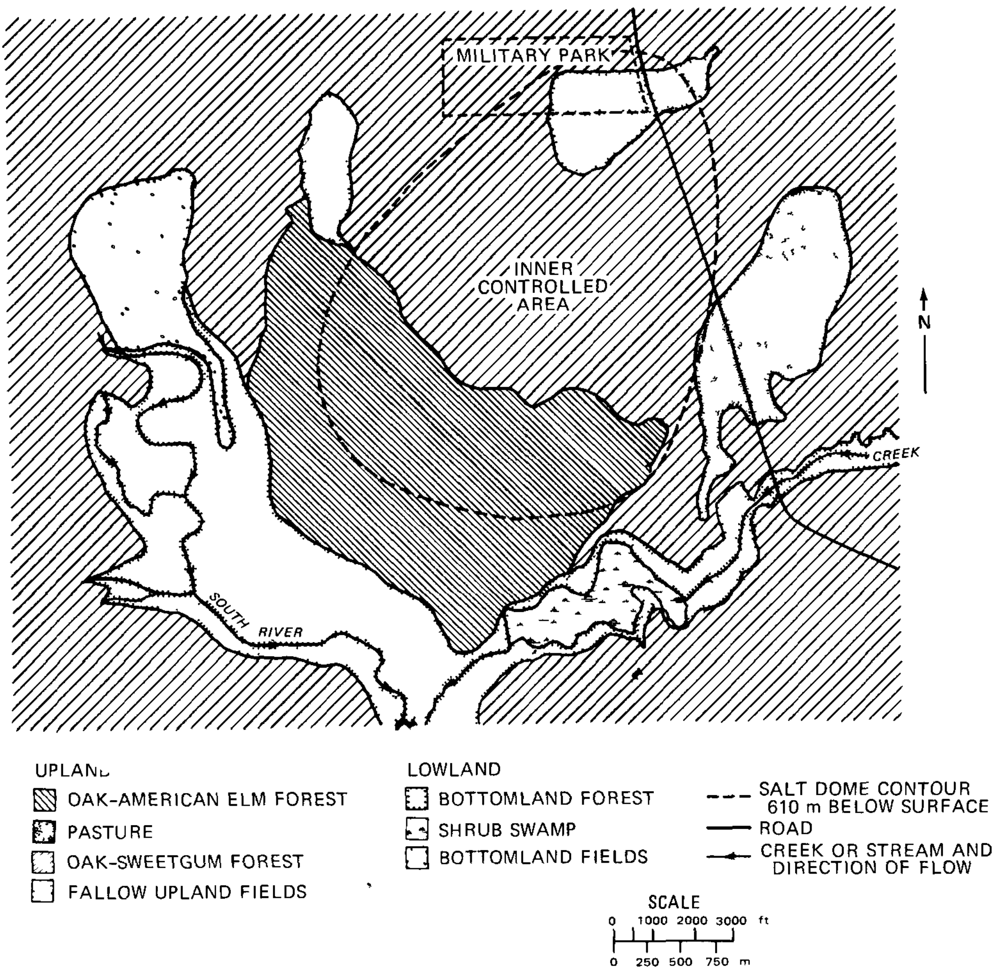

Figure 2 Vegetation types on and in the vicinity of the reference site.

mississipiensis), which was assumed to inhabit the shrub swamp.

Similarly, a demographic and socieoeconomic setting typical of rural areas in the Gulf interior was developed. The setting included a nearby small town of about 1500 people, other nearby small rural areas, and several homes and a historical site (Military Park) on the repository site. Larger towns with populations of 25,000 or more were located approximately 32 to 96 km (20 to 60 miles) from the site.

Thus the hypothetical area created was representative of realistic conditions. The site was assumed to comply with existing NWTS siting criteria for a nuclear waste repository (Office of Nuclear Waste Isolation, 1980), as well as with restrictive regulatory requirements such as Executive Order 11988 (May 24, 1977), which requires avoidance of floodplains; U. S. Department of Agriculture 101(b)4, which protects prime farmland; the Wilderness Act (United States Code, 1974) and the Wild and Scenic Rivers

Act (United States Code, 1968), which protect areas of special recreational use; and the Endangered Species Act (United States Code, 1973), which protects critical habitat.

This characterization was only an example of the kinds of data contained in an environmental report (ER); more detailed information based on literature, laboratory, and field surveys would be required in an ER submittal to the NRC. The types of data needed and the programs that would be conducted to obtain necessary data for a licensing document were identified. In addition, statistical tests or comparisons required to demonstrate regional stability, long-term trends, or local variation for specific characteristics were identified, and their uses were described. On the basis of present understanding of data requirements for repository impact evaluation, existing data-gathering techniques are sufficient to provide information for evaluating short-term impacts.

Reference Repository. A repository design, based on the Conceptual Reference Repository Description (CRRD) (Bechtel National, Inc., 1979), was developed. Figure 3 shows the layout of the facility and the site area, including the boundaries of the inner controlled area and site exclusion area and the site security area. The exclusion area, which contains the operating and processing facilities, covers an area of approximately 162 ha (400 acres) and is surrounded by a chain link security fence. Located within the exclusion area is the site security area, which contains the surface facilities for operating the repository and is surrounded by a double security fence. The inner controlled area, which is owned in fee-simple by the the applicant, approximately matches the dome contour at a depth of 640 m (2100 ft), the repository depth. The applicant controls surface and subsurface uses within this area, as well as access roads and railroad spurs. The outer controlled area is a 3.2-km (2-mi) annulus around the inner controlled area. The applicant controls limited subsurface use within this area, but compatible surface uses are permitted.

Within the exclusion area, surface facilities include the waste-handling building, administration and operations buildings (including fire, security, and medical services and maintenance, storage, and laboratory facilities), excavated salt storage piles, wastewater ponds, a railroad spur, and roadways to serve the various facilities. Figure 4 shows a perspective drawing of the surface facilities.

Underground development would be 640 m (2100 ft) below the surface. The repository would be 556 ha (1374 acres) in area, with an additional buffer-zone annulus of 245 m (800 ft), for a total of approximately 850 ha (2100 acres).

Access to the underground areas would be through five shafts: the disposal exhaust shaft, the development exhaust shaft, the men-and-materials shaft, the waste shaft, and the ventilation-supply and emergency-egress shaft. The repository would be developed from seven main access corridors branching out from the shaft pillar area. The corridor extremities would be developed first, and, after the period when retrieval is possible, these would be backfilled with excavated salt. Development would proceed simultaneously with backfilling, and rooms near the repository center would be excavated as exterior rooms are filled.

Regulatory Requirements

Preliminary assessment indicates that environmental effects are within an acceptable range, provided the design of such systems as the coal-fired steam generator, mined-rock

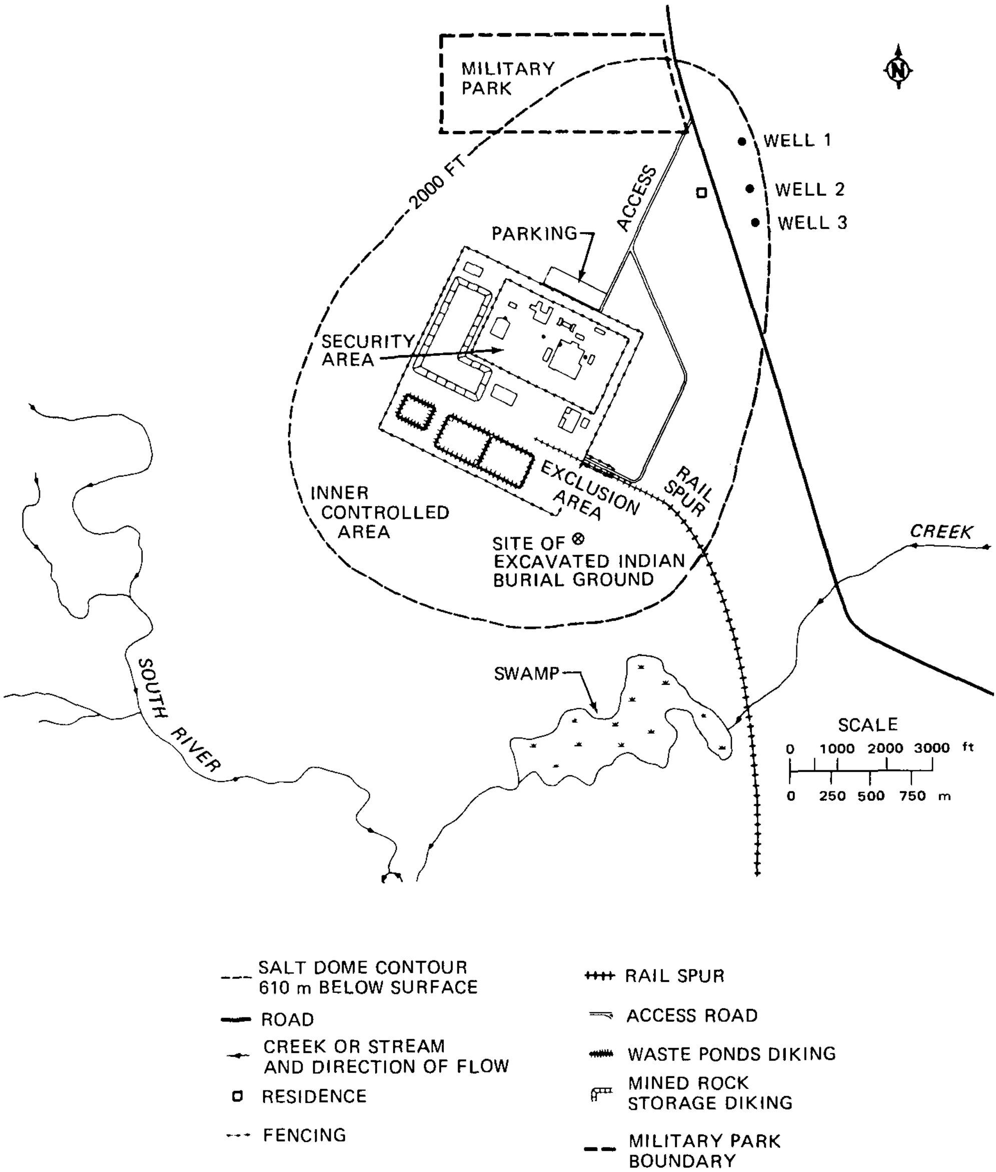

Figure 3 Layout of reference repository site.

handling facilities, and wastewater treatment incorporates mitigation of environmental effects. The reference repository design used in the evaluation would comply with existing air, water, and solid waste regulatory requirements. Such regulations are continuously being updated, however, either through additional legislation or interpretation by the courts, and will require constant surveillance to assure compliance.

Compliance with National Environmental Policy Act (United States Code, 1969) and Council on Environmental Quality (CEQ) (1978) require-

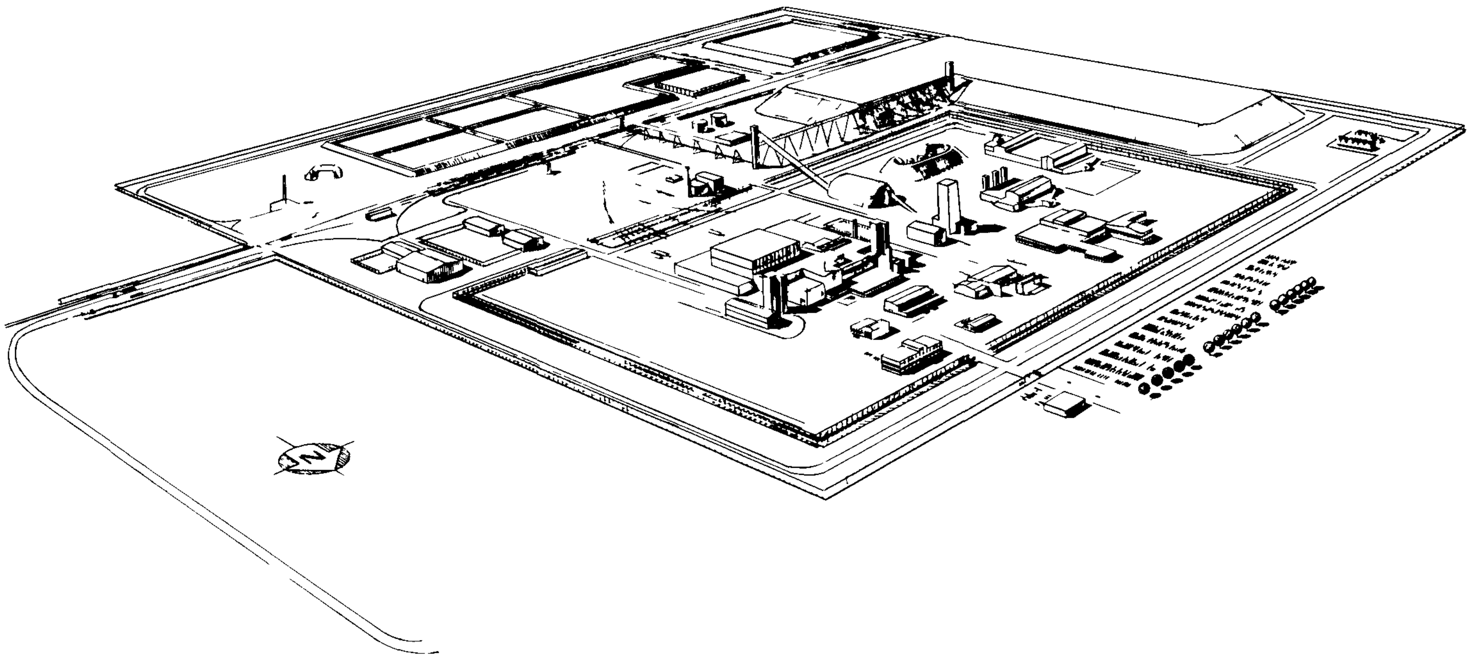

Figure 4 Perspective of repository surface facilities.

ments are the responsibility of the NRC. In response to new CEQ guidelines issued in November 1978, the ER guidelines for all nuclear facilities may be revised. Such revisions are reflected in draft ER guidelines for a geologic repository (Nuclear Regulatory Commission, 1978). The analysis was based on these guidelines, as well as on existing ER guidelines for other nuclear facilities licensed by NRC.

Short-Term Environmental Effects

As previously stated, the short-term phase encompasses facility construction, operation, and decommissioning. Generally, local and regional effects during this phase will be similar to those of any large industrial development in a rural area.

Local Effects During Construction. During construction, potential local environmental effects include:

- Temporary effects of construction activities on soils, vegetation, air quality, water quality, noise, and aesthetic quality on site
- Preemption of surface and subsurface resources unless uses are compatible with repository integrity
- Excavation of large quantities of salt, which must be either stored on-site or acceptably disposed of off-site; related air, water, soil, and ecological impacts caused by accumulation of blown or leached salts
- Ground and surface water pollution from dust, blown salt, spills, and other air or soil pollutants that may percolate through or run off the soils
- Consumption of water for construction processes
- Growth of population in small towns within an 8-km (5-mi) radius by as much as 60%, and subsequent increased demand for services, infrastructure, and housing
- Change in the character of the area from rural agricultural to industrial and commercial as a result of in-migration of skilled labor and white-collar workers
- Increased local traffic, with potentially increased road maintenance, congestion, and highway accidents

Regional Effects During Construction [Within an 80-km (50-mi) Radius]. Potential regional effects during construction include:

- Air pollution from fugitive dust, vehicle emissions, and blown salt from transportation of workers, materials, and the excavated salt
- Increase in regional employment and income by 2%
- Increase in regional population and demand for housing and services by 1%

Local Effects During Operation. Potential local effects during operation include:

- Small releases of radioactive gases and particulates during waste handling within limits set by government regulations (Code of Federal Regulations, 1960)
- Air pollution caused by vehicles, repository equipment, and the steam-generation facility during operation
- Solid wastes generated by the air pollution control system, the coal-fired steam-generation facility, and the sanitary facility
- Growth of income in small towns in the area through the operations phase
- Groundwater and surface-water pollution from dust, blown salt, spills, and other air or soil pollutants that may percolate through or run off the soils
- Consumption of water for operation processes

Regional Effects During Operation [Within 80-km (50-mi) Radius]. Potential regional effects during operation include:

- Air pollution from fugitive dust, vehicle and steam-generation-facility emissions, and blown salt
- Potentially adverse effects from transportation of the high-level radioactive waste and salt by rail or truck
- Added costs for maintenance of roadways for increased commuter traffic and population associated with economic growth

Local Effects During Decommissioning. Potential effects during the decommissioning phase will be similar to those during construction and will include:

- Small effects of decommissioning activities on soils, vegetation, air quality, water quality, noise, and aesthetic quality on-site
- Air, water, soil, and ecological impacts caused by accumulation of blown or leached salts during disposal of excess salt
- Consumption of water for decommissioning processes
- Loss of population, unemployment, loss of income in local small towns by as much as 40 to 50%, and subsequent local economic depression, reduced sales, excess housing
- Reversion in the character of the area from industrial-commercial to rural agricultural because of emigration of skilled labor and white-collar workers (assuming other industries do not develop significantly during the 30-yr operating life of the repository)
- Reduced local traffic
- Reduced demand for local services, schools, and recreational facilities
- Small releases of radioactive particulates during decommissioning of contaminated facilities, which are controlled within limits established by government regulations

Radiation exposure from the heavily shielded transportation routes will not be significantly greater than background radiation levels.

Regional Effects During Decommissioning [Within 80-km (50-mi) Radius]. Potential regional effects during decommissioning include:

- Air pollution from fugitive dust, vehicle emissions, and blown salt
- Groundwater and surface-water pollution from dust, blown salt, spills on site, and other air or soil pollutants that may percolate through or run off the soils
- Decreased regional employment and income, regional population, and demand for services

The major effects of construction, operation, and decommissioning will be local and potentially significant. For the reference site, the most significant adverse effects are expected to result from:

- Handling, storage, and disposal of mined rock (salt)
- Socioeconomic changes in the rural communities

Adverse effects of handling, storage, and disposal of over 38 billion tons of mined rock were considered to be the major environmental concern under normal (nonaccident) conditions, particularly in regard to salt as the geologic medium. Salt (NaCl), except when used in small amounts, as for culinary purposes, is toxic, particularly to vegetation and aquatic organisms. Potential impacts from windblown salt deposition and stockpile runoff can be mitigated by currently available technology, e.g., covered conveyors, dust control, catchment basins, holding ponds, and impermeable liners. Little quantitative research has been done on the effects of windblown salt on biota, soils, and surface-water and groundwater quality; however, on the basis of preliminary dispersion analyses, the most severely affected area would probably be within the boundaries of the exclusion area (within 100 m).

Socioeconomic effects will derive from the influx of a large construction and operation labor force, which will increase local income and related economic factors but will also create a demand for services that cannot be readily supplied during the expanding years. Conversely, during decommissioning, the reduction in operating labor force could create a locally depressed economy as demands for housing, services, and retail products are reduced. Federal and state programs to compensate local communities (payment in lieu of taxes) for the increased demand during expansion and decreased economic activity during contraction of the labor force have not been defined at this time.

Major environmental concerns related to radioactivity include:

- Protecting the work force and the public from radiation exposures during normal operations
- Protecting the public from exposures during transportation of the high-level radioactive waste to the repository
- Protecting the work force and the public from exposure during accidents

On the basis of available models and present understanding of repository operations, potential exposure of workers to radiation and potential releases of radiation to the surrounding environment during normal operations are expected to be well within existing standards.

Truck and rail routes will be selected to minimize population exposure by avoiding urban and suburban areas whenever possible. Regulations proposed by the Department of Transportation would control the shipment of radioactive waste by requiring that heavily populated areas be avoided, transit times be minimized, drivers be properly trained, and route plans be filed (Nuclear Regulation Reports, 1980). Other potential mitigating measures include design of transport shield containers that are extremely resistant to damage and payment for road and railroad upgrading, maintenance, and traffic control along waste transport routes.

Accidents or abnormal events may occur during repository construction, operation, and decommissioning as a result of natural phenomena or facility malfunctions. Plausible, though unlikely, accidents involving radioactivity could result from design-basis events—e.g., the "dropping" of a canister of waste down a shaft or a waste-handling incident. The analyses of potential design-basis accidents indicated that no unacceptable off-site releases would be caused. The safety of all radiation handling systems will have to be demonstrated before the repository can be licensed.

The most severe credible accident not involving radioactivity is the possibility of a direct hit on the salt storage area by a tornado (probability, 1 in 7000). Such an accident would result in salt dispersion to distances of 10 to 60 km (6 to 37 mi) downwind from the repository.

Long-Term Environmental Effects

Because of the long time period of the isolation phase—thousands of years—evaluation of environmental effects during this phase presented a unique challenge. Traditional methodologies for impact analysis require that baseline conditions be defined. Environmental baseline conditions cannot be defined for extended time periods (beyond 40 to 50 yr), however, because of the unpredictability of human sociological, economic, technical, and political development. The range of possible socioeconomic conditions is almost boundless. It was decided, therefore, to assess potential impacts caused by the isolated waste using existing baseline conditions. Environmental effects of isolation under normal conditions were evaluated, as were effects caused by unusual conditions that result in release of radionuclides to the biosphere.

If the disposal system performs as expected, the radioactive waste emplaced in it will remain isolated from the biosphere for hundreds of thousands to millions of years. This long-term containment and isolation will be provided by the multiple barriers of the waste package, the repository structure, and the site. In the absence of human interference and other unexpected phenomena, the repository will serve its containment and isolation function while the radioactivity of the waste is diminished by decay. Potential environmental effects of this normal condition will be limited to heat emanating from the emplaced waste and subsidence resulting from compaction of the backfill material surrounding the waste canisters. Thus, environmental effects may include:

- Surface subsidence in 100 yr, which will be partially offset by thermal surface uplift during the same time period
- An increase in surface soil temperature after 6000 yr
- Potential increases in temperature of groundwater and groundwater-fed surface waters
- Long-term restrictions on exploitation of deep mineral resources on and near the site
- Possible long-term restrictions on land use at the repository

Unusual events that result in radioactive releases may be induced by natural phenomena, by human interference, or by the waste itself. Models to be used in calculating releases are being developed. Phenomena were analyzed according to the mechanisms by which releases would be made to the biosphere. Figure 5 shows potential pathways to the biosphere.

The following conceptual scenarios were identified as mechanisms for releases to the biological environment:

- Direct connection to the surface through a single-flow pathway, such as a drilled core
- Circulatory flow to the surface (down one pathway, out another) through two or more connections, such as drilling to an aquifer that surrounds the repository
- Direct uncovering of the repository to the atmosphere
- Flow from one aquifer, through the repository, to another aquifer, then to a discharge point of the receiving aquifer
- Releases to an aquifer, with discharge to the biosphere at a well

The processes that may cause releases of radioactivity through one or more pathways include (1) waste- or repository-induced phenomena in the near field (the immediate environs of the site), (2) natural phenomena, and (3) human interference. These phenomena were analyzed in terms of their likelihood of occurrence, and calculated releases for such phenomena were presented only as examples of applications of performance assessment technology. Results of these analyses indicate that the vast majority of possible disposal-system conditions would not deliver any measurable doses to people. Some unlikely, though possible, phenomena (e.g., groundwater flow directly through the repository) could deliver

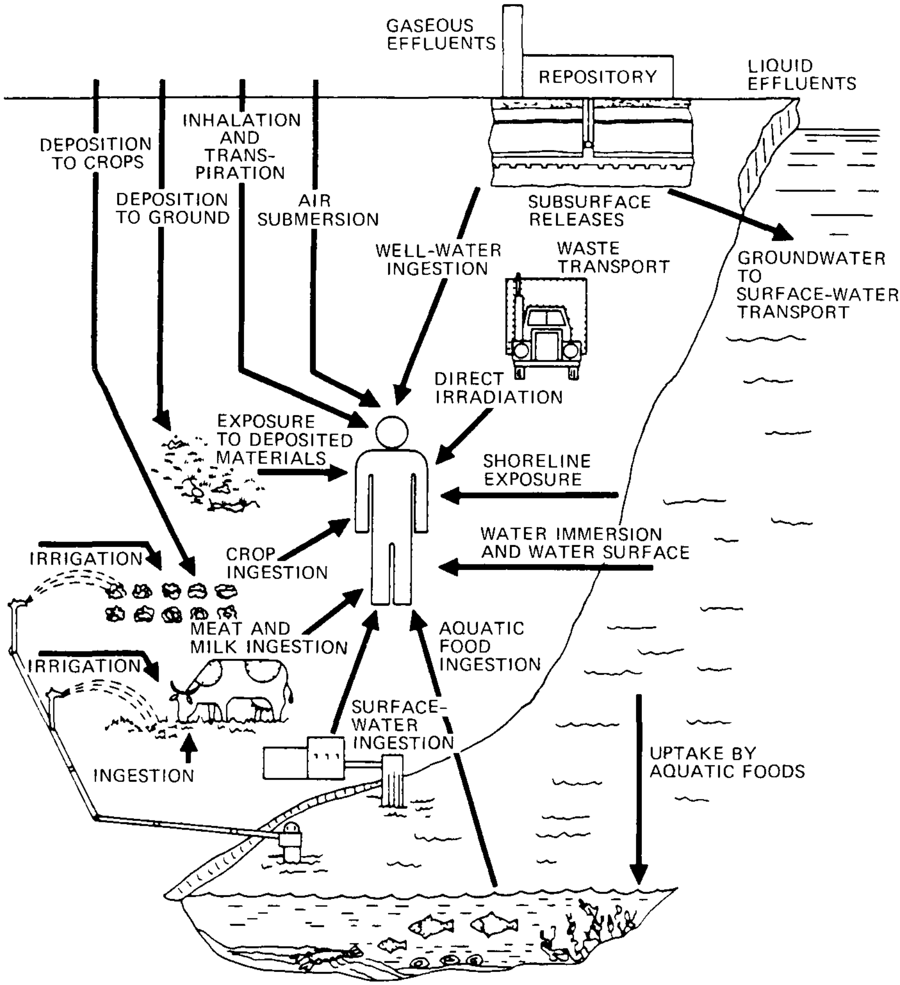

Figure 5 Pathway parameters relevant to radiological dose calculations.

radiation doses that would be a fraction of those delivered by natural background radiation (Department of Energy, 1980).

Major environmental issues related to long-term effects include:

- The need to better understand the biologic response to small temperature changes in soil, groundwater, and surface water
- The need for methods to reduce the risk of human interference with the repository over the long time period of its service (up to 10,000 yr)

The biological effects of small temperature increases in soils and water resources have not been studied for this analysis. Presumably, the design basis for the heat loading of the high-level radioactive waste canisters will take into account potential effects of heat on biota. Possibly, examples exist from nature and from thermal effluent studies performed for major industrial projects. Such data have not been compiled and analyzed with regard to the potential repository effects, however.

With regard to human interference with repository performance, a special NWTS task force has been created under the ONWI Licensing Program Office to study this issue to identify methods of reducing the risk of human interference.

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Review of Institutional and Socio-economic Issues for Radioactive Waste Repository Siting

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The early developers of nuclear power had three failings—they knew too much about radioactivity, not enough about geology, and almost nothing at all about dealing with the public and its reactions.
Hammond, 1979

In recent years, the importance of social and institutional issues in the siting of radioactive waste management repositories has been recognized. This study deals with the possibility of using incentives to assist in siting repositories and outlines some of their uses, limitations, and preconditions. Limited survey data and other studies indicate that incentives may help encourage people to formulate positive positions on radioactive waste repositories. In an overall siting strategy, incentives are just one part of a structured process involving the creation of a mutually acceptable set of arrangements that make certain guarantees and confer certain benefits in exchange for the acceptance of the proposed facility. Because the range of needs to be fulfilled is varied, a package of incentives is likely to be more acceptable than any one single incentive.

The purpose of incentives is to encourage local approval by minimizing and redressing costs and providing missing benefits. Most previous discussions of incentives have emphasized mitigation mechanisms only. This paper also identifies compensation, incentives, and criteria by which compensation or an incentive system can be evaluated. The study provides the means by which incentives can be identified, assessed, negotiated, and implemented by affected parties and attempts to show where incentives fit into an overall siting strategy by developing a classification scheme and an analytical framework that capture: (1) the *preconditions* that must exist before any incentive system can be considered; (2) the *"objective" features* of an incentive, such as adequacy and ease of administration; (3) *community perceptions* of an incentive, such as interpretability and relevance; and (4) the *consequences* of implementing an incentive, such as distributional effects and conflict and consensus.

Social and Institutional Dimensions of Radioactive Waste Management Policy

The future of nuclear power as an energy supply option in the United States is predicated in large part on resolution of public acceptance problems related to the storage and disposal of waste products. Evidence from at least one poll suggests that solving the waste problem would increase support for the construction of nuclear power plants by 15% (to around 60% support), and perhaps by as much as 25% (to about 70%).*

In recent years the importance of social and institutional issues in siting radioactive waste management repositories has been recognized. Within this subset of issues, the siting of repositories over the objections of members of potential host communities is viewed as especially problematic. The extent of the problem is illustrated by the Wisconsin community survey, which reported that two-thirds of the respondents were strongly opposed to siting a repository in their community, and just 5% strongly favored a waste repository.

The range of social and institutional issues that currently plague governmental attempts to resolve radioactive waste management problems is extensive. Kasperson (1980) clustered them into institutional uncertainties, which include institutional fragmentation, gaps in regulatory authority, waste management program inconsistencies, intergovernmental conflict, and a diminution of institutional credibility (see also Abrams and Primack, 1980). The lack of institutional credibility was illustrated in the previously cited survey data in which utilities, state government, and the federal government (institutions that have traditionally been sources of much of the information about nuclear power) were not perceived to be reliable sources of information about nuclear waste. Only 6 to 9% of the respondents believed any of these institutions to be the most reliable information source.

**The poll, which was directed at ascertaining the opinions of local residents about the siting of nuclear waste repositories, was conducted in 1980 by John Kelly, Complex Systems Group, University of New Hampshire. There were 426 respondents in three small Wisconsin communities. Comparisons with other surveys on nuclear power suggest that these response patterns are fairly typical.*

To this list of institutional uncertainties, Kevin (1980) added systemic and idiosyncratic characteristics—emerging trends in federalism (e.g., the sagebrush rebellion), perceptions of incompetent institutional performance on nonwaste management issues (e.g., virtually all institutions after the Three Mile Island accident), and increasing awareness of serious deficiencies in nonradioactive hazardous waste management (e.g., at Love Canal and Toone; see also Carnes, 1982).

Public acceptance in potential host communities is questionable considering these social and institutional difficulties. Yet public acceptance of such a facility is necessary to site, construct, and operate the facility for long periods of time. If people fear hazardous waste to the extent that they believe it represents a real risk to their health and well-being, a disposal facility is unlikely to be sited in their community through any mechanism (Bacow, 1980). The major question facing facility sponsors can be reduced to how to develop and maintain local constituencies in host communities for the repositories.

Issues center on problems of spatial equity, intergenerational equity, and what Kasperson (1980) characterized as the labor-laity equity problem (i.e., impacts on waste management workers vs. the general public). Local citizens often indicate strongly that those who generate radioactive waste should be responsible for it. In the Wisconsin survey, nearly 70% of the sample said that a waste repository in Wisconsin should be only for wastes generated in Wisconsin. Less than 7% were willing to accept wastes from the region (Midwest), and only 5% were willing to accept wastes from anywhere in the United States. These general trends have been borne out in other studies (Kevin, 1980).

All the issues have combined to create a stalemate, both in terms of making national policy on siting and other social and institutional problems and in terms of how a national siting policy might be implemented at the state and local level. The stalemate can be broadly characterized as a lack of public acceptance of proposed radioactive waste management repositories, together with general uncertainty about the acceptability of nuclear energy (including wastes).

The United States has already generated considerable quantities of radioactive waste. Even if a decision were made to halt immediately all activities that generate radioactive waste, the current accumulation requires safe and secure storage and disposal. Thus it is unavoidable that certain communities and their citizens will eventually host radioactive waste repositories.

A number of recent studies and policy initiatives have suggested that diverse incentives can be used to address the costs and risks to increase local support and offset local opposition to repositories in potential host communities. Incentives are judged to be preferable to disincentives (e.g., federal or state preemption) since incentives may generate support, whereas disincentives do not eliminate opposition (Bacow, 1980). Suggested incentives have included private insurance (Goetze, 1981), rebates on electric utility costs (Starr, 1980), payments in lieu of taxes (Bjornstad and Goss, 1981), and a variety of waste management program guarantees designed to respond to the concerns of state and local governments (Kevin, 1980). Rarely, however, have incentives been systematically identified, investigated, or evaluated.

Most communities are willing to accept and many actively seek new economic activities, anticipating that the benefits from new jobs, additional tax dollars, and the general vitality associated with new enterprise will outweigh the associated social, economic, political, health, and environmental costs or adverse impacts. When communities reject such activities, they do so because they perceive a benefit-cost-risk imbalance. When costs and risks go uncompensated, communities are reluctant to host facilities, and, when they are exactly compensated, they view the facility with indifference. Only if they view the benefits as outweighing the costs and risks will they have an interest in hosting the facility (Bjornstad and Johnson, 1981). If they lack information and have uncertainties regarding the benefits, costs, and risks associated with new developments, communities cannot make optimal decisions.

Local reactions to existing defense facilities involving major radioactive waste handling and storage activities suggest the key role of the local benefit-cost balance. For instance, interviews in the impact area surrounding the Savannah River Plant in South Carolina revealed primarily favorable reactions in areas where substantial employment of local residents occurred and significantly less favorable attitudes and more concern about health effects in a county that received minimal employment benefits (Department of Energy, 1981).

The nature of the waste repository requires dealing with benefits and costs at a level of detail exceeding that necessary for typical industrial enterprises. The differences that distinguish the waste facility are quite familiar; e.g., such a facility handles potentially hazardous materials; its security must be maintained over long time periods; and the federal government is expected to own the facility. Each characteristic leads to particular costs and lessens the likelihood of normal benefit structures.

There are at least five types of costs associated with repositories—

infrastructure impacts, chronic damage, accident potential, opportunity costs, and extraordinary costs. It is noteworthy that only the first two are conventionally assessable as reimbursables, a circumstance that has led government and industry representatives to focus on them. Citizens in affected communities usually focus their attention on the latter three, however. First, the infrastructure impacts are stresses placed on community service delivery systems when development occurs. Second, chronic damage to physical systems occurs, e.g., pollution from normal operating conditions of any large facility. Third, there may be perceived risk to physical systems, as, e.g., in the event of a large nuclear release. Fourth, there are possible opportunity costs—the unrealized benefits a community must forego to host a waste facility (e.g., the tax receipts from a privately owned industrial activity). Finally, there are extraordinary costs; these include those resulting from such factors as stigma and uncertainty, among others.

From the local perspective, the benefits of radioactive waste repositories are largely absent or are difficult to identify. The jobs provided by construction and operation are the principal benefits, but many construction jobs will be filled by nonresidents. The taxes that the community would derive from most private industrial facilities will not be paid if the federal government is the owner.* The outcomes of efforts to tax federal contractors, as for example the federal facilities in Oak Ridge, TN, are as yet unresolved (as evidenced in issues of the newspaper *The Oak Ridger* after 1980, particularly June 15, 1981). Furthermore, the presence of the federally owned facility may prevent or forestall receipt of other possible future benefits for the community by preventing different and more beneficial uses of the land by tax-paying private industries.

The fact that sitings of hazardous-noxious waste facilities provide regional and national benefits is of little interest to local people, for whom the costs loom large, and in many cases the local benefits are seen as insufficient. The National Governors Association (1981) made a policy statement on siting hazardous waste which recognized this frequent combination of a widespread dispersion of benefits and geographic concentration of costs. The local response invariably recognizes the need for such facilities but asserts that "somewhere else is better" (O'Hare, 1977; Peelle, 1980).

Incentives for What? A Classification Scheme

Recent discussions of incentives for encouraging the acceptance of noxious facilities have emphasized a fairly narrow approach by focusing on the use of direct payments. Incentives, however, should be viewed as a much broader range of actions that may promote acceptance. Nonmonetary incentives may play an equally important role with economic ones in the siting process. Packages or mixes of incentives may be more attractive than any single benefit mechanism. This section identifies and defines a variety of different types of incentives, presents an overview of a range of options within each type, and provides examples and illustrations of how they might be used in siting radioactive waste facilities.

Incentive systems must be distinguished according to their purported function. Incentives can provide:

1. Mitigation against potential problems resulting from normal sit-

**Unless special arrangements are made, enabling legislation is passed, and financial authorizations are actually implemented through the budget process.*

ing, construction, and operation of the facility

2. Compensation for real and perceived costs incurred in the event of an accident or anomaly

3. Rewards for the local community for assuming risks and costs to meet nonlocal (i.e., national, state, regional, and international) needs

It is important to distinguish incentive types so that we can determine why a particular incentive might be offered, to whom it might be offered, and what institutional and administrative arrangements might be necessary to implement it. Distinguishing incentive types also facilitates the development of evaluative criteria and identification of the criteria that are particularly relevant to each type of incentive.

Among the three types of incentives mentioned, compensation and mitigation have become legitimate means to use in alleviating many of the direct, quantifiable impacts of facility siting. For example, Resolution 5-4 of the State Planning Council (SPC) on Radioactive Waste Management (1981) reads:

> The State Planning Council recommends that as the Federal government has the responsibility for developing repositories it should accept the responsibility for socioeconomic impacts resulting from such repository development. Impacts should be identified early in the repository development process, and be independently assessed by state/tribal/local governments with Federal funding assistance prior to a DOE application for a construction license. After the NRC decision to license repository construction has been made, Federal government impact payments should be made to state or tribes to distribute, in accordance with impact experienced, to affected jurisdictions.

However, the SPC "did not specify what types of impacts merit compensation but concluded that only *quantifiable* impacts should qualify, and not impacts caused by perceived risks of a high-level waste repository" (State Planning Council on Radioactive Waste Management, 1981). This constraint may be extremely problematic because of a large differential between public and technical perceptions of risk associated with nuclear facilities (Slovic, Fischhoff, and Lichtenstein, 1980). The reward type of incentive has had more limited application, but it includes a broad range of options that have been applied by private and governmental entities in past siting practices. This incentive assumes a precondition that health, safety, and personal rights of interested parties are protected. The interactive and currently casual use of these terms and concepts has tended to blur the distinctions between them.

The remainder of this section discusses each of the incentive types in some detail and identifies possible options within each type. These options are not meant to be considered as exhaustive but merely represent a variety of potential mechanisms to mitigate, compensate, or reward the local community for its willingness to host a radioactive waste repository.

Mitigation. Mitigation is defined here in a slightly narrower framework than is generally the case. It encompasses only actions to alleviate the potential risks or anticipated negative impacts that could occur during *normal* construction and operation of the facility. Mitigation alone is not likely to neutralize local opposition to waste facility siting since it is not possible to eliminate all risks and local costs completely. It is, however, an important component of a comprehensive, multipurpose incentive package. Mitigation is based largely on the *perceptions* of the local population, and successful mitigation requires public involvement in determining the buffering, monitoring, and other options needed and in administering these options (Bacow, 1980). Comprehensive planning by potential host communities is also necessary to enhance the potential of mitigation strategies. There are a variety of mitigation measures and mechanisms.

Buffers and Land-Use Management. Money could be provided to purchase land surrounding a facility to a certain distance to prevent human occupation of potentially hazardous sites. Development rights or easements could also be employed. Many state hazardous-waste-management laws contain provisions for land management. For example, Indiana requires owners to place restrictions on their deeds prohibiting land disturbance after closure; Kansas requires land to be owned by the repository operator in "fee simple" (i.e., absolutely and unconditionally); and Michigan has provisions to restrict future uses of sites without permission.

Monitoring and Detection. Mechanisms for monitoring for potential hazards to residents of potential host communities could be used to alleviate local fears or anxieties and to alert people to a problem if it should occur. Radiation detectors would assure people that they are safe and enable officials to detect potential problems. Many states with hazardous-waste-management programs require mandatory safety inspection of facilities on a regular basis. Given the problems of institutional credibility, independent monitoring by the community may be required.

Emergency Response—Preparedness. Personnel, equipment, and information could be provided to the community to respond to a problem if one should occur.

Public Education. A program could be established to teach the public about disposal facilities and their safety design. This could be an integral part of many incentive systems.

Socioeconomic Impact Mitigation. Before the siting of a facility, plans could be developed and advance payments could be made to prevent negative socioeconomic impacts associated with facility construction and operation. A variety of ways to implement mitigation programs are feasible (see the section entitled Direct Payments).

Land Value Guarantees. Land and property values could be guaranteed against a real decline in value as a result of the facility. This mechanism could include a program where people wishing to relocate would dedicate their property to the facility and/or the government and would be paid the fair market value.

Compensation. Definitions of compensation always include the notion of loss or defect, which is repaired, replaced, or otherwise recompensed (see, for example, *The American Heritage Dictionary*). The "counterbalancing" of a lack or a loss is another common thread in more specialized definitions. O'Connor (1980) indicated that the traditional view of compensation involves the concept of "making-whole" or providing replacement costs. This implies that people can be reimbursed for social costs, which is true only to the extent that it is possible to recreate the status quo. Some changes in local conditions are quantifiable but are not directly traceable to waste facility siting; these impacts are compensable in theory but not in practice because of the difficulty in developing a precise compensation scheme (Bacow, 1980).

At least four types of compensation—trust funds, insurance schemes, guarantees, and contracts—can be defined.

Trust Funds. Money could be contributed by the government, industry, or both, in a lump sum or a yearly contribution, which would accrue interest.

Difficulty could exist in establishing the amount of money that should be placed into the fund. This stems chiefly from uncertainties and disagreements over the risks. Furthermore, the process by which compensation would be awarded and administered is likely to be complex.

For example, Florida's Recovery and Management Act calls for the establishment of a Hazardous Waste Management Trust Fund to finance emergency actions and other needs. This fund is established through a 4% excise tax on disposal until the accrual reaches $30,000,000 and 2% thereafter. Other states have or are developing similar programs. The extent to which these funds could be used to compensate for damages is unclear.

Insurance Scheme. The federal government or a private company could create an insurance pool in which all or part of the premiums would be paid by government and/or industry. Claims for damages could be filed against this pool.

Precedents for federal involvement in insurance for low-probability/high-risk events include the Price Anderson Act and the National Flood Insurance Program. A significant problem with an insurance scheme is that it may cause people to suspect that something negative is likely to occur, although there is some evidence of more positive reactions to such schemes (Kunreuther, 1979). Some states, e.g., Kansas and Oregon, are adopting legislation for hazardous waste facilities which requires owners to possess liability insurance.

Currently it would be difficult to set the appropriate minimum aggregate coverage on liability for accidents at radioactive waste storage facilities. This problem is highly evident for hazardous waste management as well (Wolf, 1980).

Assumption of Liability. The federal government and/or industry could provide written assurance to assume liability to a certain level of damages from an accident or anomaly. Legal action could be taken to recoup losses. Most states that have developed statutes on hazardous waste management require the developer to assume liability for environmental damage and adverse health and safety impacts.

Contracts. The federal government and/or industry could enter into a contract with a local government to compensate for accidents or anomalies.

Rewards. Rewards can be used to induce communities to accept concentrated costs and risks, such as changes in community character and the stigma presently associated with garbage disposal or "dumping" activities. Assisting in solving national problems by assuming additional burdens is not very compelling to local populations (Brenner, 1979). Rewards may take many forms and may be negotiated with individual communities. These must be perceived as positive inducements, however, and not as payment for actual damage or a form of subscription where the poor of the land are paid for their willingness to accept the refuse of their richer neighbor. If perceived as either of these, the reward will be viewed as a bribe. A broad variety of rewards have been identified.

Direct Payments. Single or yearly payments could be made to communities or to families and individuals residing within the community. A variety of implementation schemes to accomplish this are available. In addition, the level of the grant can be determined in a number of different ways.

Potential schemes so far include government grants, user fees, revenue sharing, gross receipt taxes, and waste surcharges. For example, Indiana's Hazardous Waste Facility Site Authority Act calls for a host county to receive $50/ton of hazardous waste disposed of in or on the land. Recently it has been suggested that a tax on the nuclear industry would enable grants to communities of $5000 per household or a direct grant to families of a like amount (*Radioactive Waste Management,* 1981). Georgia has proposed a 1% gross receipt tax on hazardous waste. Kentucky authorizes counties to collect license fees on waste facilities. Ohio has authorized the expenditure of $500,000 over 3 yr in the form of local grants to encourage the siting of hazardous waste facilities. The National Governors Association (1980) recommended a special congressional discretionary fund to provide benefits to state and local governments to promote acceptance of low-level nuclear waste disposal sites.

If the facility could make payments in lieu of taxes, the transfer would occur directly between the facility and the community. A variety of other transfers of this type are also possible. These offer the advantage that, since they are assessable to the facility, facility users will bear their burden. Other transfer payments could occur between the state and the community or the federal government and the community. Again a variety of options are available; these range from such general purpose funds as general or special purpose revenue sharing, planning grants, and special project grants to specific waste facility impact grants. These payments offer the disadvantage that funds are paid out of the general revenues of some other government entity and may not be borne by facility users.

Bonus Community Services. Funds could be provided to support services that are *not* required as part of a mitigation scheme, e.g., job training programs, scholarship funds, such amenities as parks or cultural functions, or additions to essential services such as schools, police and fire protection, and public works. Services may be provided directly by either the facility or some higher level of government. This relieves the community of administering the service, but it may also mean that the community loses control over the quality of the service. If the service is facility related, this loss may be unimportant; but, if it pertains to the community as a whole, difficulties may arise. These services could be funded by the facility or by a higher level of government.

Tax Incentives. Tax breaks to residents of impacted communities could be one means of using the tax system to provide incentives, e.g., a state tax credit. An alternative scheme would use revenues from other incentives to replace funds collected through property or sales tax and adjust those taxes accordingly. The costs of such incentives as granting tax liability forgiveness to residents through the use of credits or deductions may be borne by the general revenues of the granting government entity or financed by taxes on radioactive wastes sent to the repository.

Advance Payments and Subsidies. Mitigation involves payment of funds to correct for negative impacts. Associated rewards might be to provide funds before they would normally be allocated or to help subsidize preimpact planning and mitigation or nonimpact related community functions by buying low-interest municipal bonds or a related scheme. Both would provide communities with added benefits; in one case, because of the early payment, the amount of the funds would be increased by the amount of the interest that would accrue. In the other, the benefit would be the savings in interest over the existing lending rates.

Infrastructure Development. Along with the facility development, infrastructure to support additional and perhaps more desirable industry and commerce could be developed. It is known, for example, that the private sector typically considers local amenities, such as schools and hospitals, as important factors in deciding where to locate an industry.

Linking. The radioactive waste facility could be "linked" in a package with other more desirable federal projects (O'Connor, 1980). Although antitrust laws concerning conditioning the sale or purchase of specific goods or services upon the purchase of another would have to be obeyed, waste repository acceptance could be made conditional upon agreement on another favorable federal or state project.

Avoidance of Other Noxious Facilities. Communities may be faced with a variety of other types of noxious facility sitings or undesirable land uses (Popper, 1981). This includes hazardous waste sites, correctional facilities, defense facilities, dams, airports, etc. Communities accepting a radioactive waste facility could be relieved of the burden of accepting other undesirables, even though current evidence in such facility siting indicates that new facilities are very often located near existing ones to minimize their political costs (the stigma already exists). Interstate and regional compacts that distribute such facilities are good examples of how this concept can be implemented (State Planning Council on Radioactive Waste Management, 1981).

A Framework for Evaluating Incentives

Criteria that can be used to evaluate the advantages and disadvantages of particular incentive mechanisms have been divided into four groups: (1) preconditions to the use of incentives in siting; (2) objective characteristics of the incentive itself; (3) community understanding of the incentive mechanism; and (4) the impacts of the incentive on the community. Figure 1 presents a simplified view of our evaluative framework, with the criteria appropriately grouped.

The preliminary criteria identified in this chapter are general, not exhaustive, and do not address the question of how the impacts of incentives might vary by site-specific or community-specific characteristics. For example, variability among communities with respect to factors such as population size, cultural diversity, home rule capabilities, geographic setting, and tax base would all likely intervene in determining the utility of a given incentive in a given community. These characteristics might be loosely identified as "host community structure" and could be incorporated as an additional variable in our framework.

We also do not address the variability of ranking or weighting of evaluative criteria by different communities and their citizens. To complicate matters even further, citizens within communities or subgroups of citizens may rank the importance of criteria differently. We should be aware of the diversity of values possible within and among potential host communities; it is not possible in this analysis to incorporate the potential effects of this diversity.

The identification, assessment, negotiation, and implementation of an incentive in a local community are complex and dynamic processes. The criteria developed in this report are also complex and dynamic, and, equally important, interactive. It is important, therefore, when evaluating an incentive to recognize that changes in the value of one criterion may affect other criteria and any overall composite score.

The purpose of these criteria is to characterize alternative incentives or incentive systems for comparison. We should note, however, that no single "best" alternative may emerge from an evaluation. By themselves, the cri-

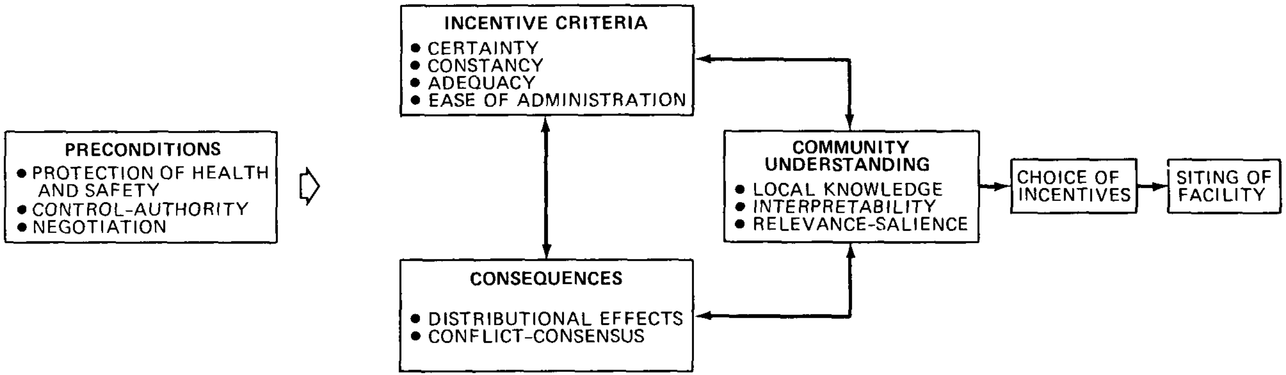

Figure 1 A framework for evaluating the utility of incentives in the siting of a repository for radioactive waste.

teria will not even indicate "goodness" or "badness." Some criteria, for example, may favor an incentive mechanism that promotes income equality, whereas others may favor an incentive that leads to greater income inequality. These are value and policy questions beyond the scope of this evaluative framework.

Preconditions for Using Incentives. The use of incentives is not a panacea for the current siting difficulties but is instead a mechanism requiring careful judgments and fine tuning to local circumstances. In the present atmosphere of suspicion, fear, and distrust of regulatory agencies and facility developers, casual attempts to offer incentives can result in public misunderstanding of their intended purpose and prompt rejection as bribes, unethical trade-offs, or unwelcome strategic ploys. The present conditions of distrust among the public must be overcome before successful and politically viable sitings can occur.

Developing a more trusting environment in which to conduct siting efforts is not an easy task nor one quickly accomplished. Groups such as the National Governors Association, the National Association of Counties, the National Conference of State Legislatures, the Atomic Industrial Forum, the Chemical Manufacturers Association, the Environmental Protection Agency, various states, and environmental interest groups are all engaged in a variety of activities attempting to address this problem. It is evident that trust is not created by the exercise of federal preemptive powers, which ignore local concerns and reject local participation (Bacow, 1980).

If local communities are to accept such facilities in a "willing trade" for benefits, local stakeholders must be involved (Haymore, 1981). Thus negotiations become a key mechanism in the process, which, if conducted successfully, is one means of creating and enhancing trust in the arrangements. Such a process gets to the heart of the siting matter—determination by the local interests of the balance between risks, costs (impacts), and benefits *which they are willing to accept.*

Preconditions identified for introducing and using incentives include:

- Safeguards for health and safety
- Control-authority arrangements
- Negotiations among affected parties

All these are requirements concerning the setting, the existing arrangements, and the context within which (1) trust can be developed, (2) costs can be identified and evaluated by local interests, (3) incentives can be suggested, and (4) bargaining can be conducted. The absence of any one of these preconditions can result in siting failure because of local demands for exorbitant levels of assurance and restitution and/or strategic withdrawal of key local interests from the siting process.

Safeguards for Health and Safety. The presence of adequate, reliable, and enforced regulations that protect the health and safety of residents near a proposed radioactive waste facility is a vital precondition for use of incentives. Much of the opposition to such facilities arises from the belief that they endanger health and safety. Both the degree and the probability of risk from normal and abnormal operation of the proposed facility may be in dispute. Existing regulations are often not known or their functioning is discounted and viewed as unreliable. The agencies responsible for enforcing existing regulations may not be respected or trusted. In these circumstances, the introduction of possible incentives to accept a facility may be received only as an illegitimate attempt to persuade people to trade their health and future well-being for certain benefits.

Through an interactive process of raising questions and concerns, reviewing and evaluating information, and comparing and testing assurances of safety, the community proceeds to determine what level of safety would be acceptable to it under what conditions. Additional guarantees, over and above those mandated by federal or state law, may be a reasonable subject for later negotiation if the need for such is identified by local interests during their examination of existing levels of protection.

Control and Authority. A local role in developing and implementing siting arrangements is another precondition for the use of incentives. Neither lack of local control nor total local control is feasible for siting noxious facilities, particularly radioactive waste repositories. If the community has no local control in the siting process, it will not be likely to be willing to host a facility; total local control would, at least initially, allow the local community to demand unreasonable sums of money and/or services from whomever provides the incentive. The end result of either no control or total control would be the same—the facility would not be sited.

What are the appropriate roles of local communities and their citizens, the state, the facility operators, the Department of Energy, and the facility users (utilities) in developing and implementing an incentive system? How much of each type of incentive (mitigation, compensation, and reward) is required? How are these values determined and by whom? When is each type of incentive implemented, and under what conditions is it implemented? Who triggers the implementation? How are disagreements on the answers to any of these questions resolved? Answers to all these questions depend on determining the proper roles of various actors in the incentive design and implementation process.

Federal, State, and Local Relations. Current federal-state negotiations (e.g., those involved in the consultation and concurrence process) and the federal-local or federal-state-local negotiations that may be required for siting repositories must be considered. Formal and informal working relationships generally already exist between federal and state entities and state and local organizations.

Between the approaches of federal preemption and state or local veto are a range of compromises in which the different levels of government share power. Though the terms consultation and cooperation (Reiser et al., 1980) and cooperative federalism (Smith, 1980) have been more widely used in the waste program in describing federal-state relations, the shared powers approach, as put forth by Kevin (1980), appears to best describe the balancing of interest among all relevant governmental units. The basic features of Kevin's shared powers system are: (1) the provision of forums for exchanging information and grievances, and (2) a checks-and-balances system that allows the state in some circumstances to halt some federal siting activities and federal power to override state objections under certain conditions. At issue in this approach are the timing and degree of specificity, types and limits of power, relative emphasis on cooperation and conflict, methods of resolving disputes, and mechanisms for expressing arrangements. The State Planning Council has considered shared-powers arrangements between federal-state governmental units. There appears to be a legitimate need for a local component in this process. Certainly there are sufficient differences in approach and needs between state and local units to require specific consideration.

Negotiation. Negotiation may be a key ingredient for a successful facility siting process. Although it is time consuming, negotiation can help balance and resolve competing interests. It is the only major public participation strategy that focuses on reconciliation of differences and, thus, has the building of consensus as a possible outcome (Susskind, 1980).

The central issue to be negotiated is under what conditions, if any, a facility can be sited in a community. Incentives enter the negotiations as a possible means to ensure a satisfactory cost-benefit ratio. Brenner (1979) affirms the view of Seidman (1980) that a policy consensus is most likely to be reached by the creation of negotiating situations among contending parties.

Most radioactive waste negotiations thus far have been conducted between only federal and state jurisdictions. In the developing field of hazardous waste management, there is an increasing number of examples of negotiation processes, including some involving incentives, which specifically incorporate local participation. The National Governors Association (NGA) (1981) outlines a negotiation approach led by local representation which narrows the agenda of items to be negotiated, generates alternatives to the proposed facility, weighs impacts, and identifies possible compensatory actions, implementation mechanisms, and guarantees. The NGA specifically recommends that funds be provided by the developer to the potential host community to enable the community to gather its own information and hire its own experts to ensure more credible and useful negotiations. The Massachusetts Hazardous Waste Facility Siting Act of 1980 is the best example of a state law with an incentive-based, site-specific negotiation approach overseen by a council representing state and local concerns (National Conference of State Legislatures, 1980). Council powers include:

- Assessing project-related social and economic impacts
- Awarding technical assistance grants to local assessment committees
- Determining compensation from developer to abutting communities
- Framing disputed issues for submission to binding arbitration between the developer and the host community

Although the need to negotiate is accepted by most participants in the siting process, it is not clear that DOE has the authority to negotiate agreements, even with states (Morris, 1980). Congressional action may be necessary to legitimize any agreements made by DOE with state and local governments. Without this legitimacy, the credibility of DOE agreements would not likely be sufficient to convince local governments to accept the facility. Even if Congress legitimized agreements between DOE and local governments, communities might still be hesitant to accept the facility because of perceived instability or inconstancy of federal decisions, whether made by the legislative or the executive branch.

Criteria Characterizing the Structure of the Incentive. A number of objective features of an incentive are relevant to the social and institutional dimensions of radioactive waste repository siting. These include its certainty, constancy, adequacy, and ease of administration. Each of these features would probably be assessed independently by potential host communities and their citizens in the process of identifying and negotiating an incentive system. Given inter- and intracommunity variation in assessing the appropriate values for these criteria, it is impossible to specify particular values that would lead to siting successes. Attention to these criteria, however, would probably lead to an earlier agreement regarding siting and incentives among affected interests than would otherwise be the case.

Certainty. Certainty refers to the likelihood that an incentive will be received or delivered as agreed. The confidence of the community that it will receive the incentive and, more generally, its confidence in the credibility of the sponsor's overall plan will be significant issues in the siting process. The degree to which incentives are perceived as empty promises will affect the level of local opposition to the incentive approach.

In general, perceptions of the sponsoring agency's credibility are closely related to the agency's expertise and trustworthiness (Krawetz, 1979). Public opinion of federal competence and responsiveness in the management of radioactive waste is often low because of a variety of factors:

- Delay by the federal government in formulating a national radioactive waste management plan (Kevin, 1980; Kasperson, 1980)
- Problems of competence and reliability, which include fears that safeguards are insufficient (Kevin, 1980), that financial controversies will arise after closure (as, e.g., at the West Valley site in New York) (Kevin, 1980), or that the waste project could be abandoned for safety or other reasons after the community had undergone significant changes and "front-end" costs (Brenner, 1979)
- Perceived insensitivity to state interests by federal agencies, including failure to keep states informed of activities within their borders and failure to uphold initial agreements between state and federal agencies (Kevin, 1980)
- Perceived alignment with the nuclear industry to provide radioactive waste management needs more than to meet national energy and environmental goals

These same issues are implicit in dealing with incentives. Delay in formulating and enacting a specific radioactive waste management plan and in demonstrating competence and reliability in safety and financial arrangements to federal-state-local governments adds to the uncertainty of the outcome. A stable funding mechanism could assure states that funds are available to remove or to maintain wastes and, where applicable, for compensation and reviewing activities. Several states have specified such guarantees in legislation dealing with the related problem of hazardous waste management, e.g., Massachusetts, Michigan, Ohio, Pennsylvania, and Tennessee (National Conference of State Legislatures, 1980).

Constancy. The constancy criterion attempts to measure the steadiness or, conversely, the variability of the incentive over time. At one extreme, an incentive may be conferred in one sum or application, and, at the other, the incentive may be continuously applied or received. For example, an incentive such as a job training program could be continuous over the lifetime of the facility. A block grant, however, might be a one-shot affair. In between, but falling toward the continuous end of the scale, are yearly impact-mitigation payments to the community, which are derived from user fees.

Constancy will be important in resolving the temporal equity concern in radioactive waste storage. Incentives that are continuous will provide a stream of benefits to the community over time. Singular schemes will favor the population present when the incentive is received. Since each extreme has positive and negative aspects, it cannot be stated which is preferable. Many studies point out that people prefer, in general, to receive benefits in the present instead of the future. On the other hand, intergenerational economic security is often judged to be an important dimension of social well-being.

Adequacy. Adequacy relates to the degree to which an incentive is sufficiently large or complete enough to make repository siting socially acceptable to a community. In view of the functions incentives are designed to fulfill, adequacy may have a number of different meanings: Is the potential compensation high enough? Are all the likely risks addressed? Does the type of incentive chosen match the perceived need?

The adequacy of the incentives and other siting arrangements is probably the key determinant of siting success from the local perspective. The process by which this determination is made, therefore, becomes a critical component and involves the preconditions for siting, control-authority, guarantees of health and safety, and negotiations.

People's perceptions of adequacy are highly variable. During implementation of a siting and incentive program, adequacy might be determined by a referendum of the population or one of the other local decision-making mechanisms referred to earlier.

Ease of Administration. Developing and applying an incentive system that is impossible, difficult, or too costly to administer does not move the facility siting process in the desired direction. Ease of administration has a number of dimensions:

1. Are procedures and institutions necessary to administer the incentive in place, or do they have to be designed and developed before implementation of the incentive (Peelle, 1980; Rochlin, 1980)?
2. Is the incentive system so complex that additional interacting layers of bureaucracy are required for administration?
3. Does the incentive system incorporate an appellate or renegotiation procedure that is burdensome, complex, and time consuming (Bjornstad and Goss, 1981)?
4. Does the incentive system allow accurate forecasting so that the receiving jurisdiction can plan and budget with confidence (Bjornstad and Goss, 1981)?
5. Are the administrative costs of implementing the incentive system disproportionately large (Bjornstad and Goss, 1981)?

Almost any proposed incentive system will probably require the design and development of some new procedures, institutions, and organizations. At the very least, the provision by the federal government of incentives to host communities will almost certainly require congressional authorization and appropriations. This may be but the tip of the iceberg, however. State governments may have to pass or amend enabling legislation for local governments; local governments may have to invent new organizations and procedures for the development, negotiation, and operation of incentives; appropriate personnel may have to be recruited and/or trained to implement the legislation; and sufficient fiscal resources may have to be provided over long periods of time to ensure continuous operation of the incentive (and the facility). Successful institutionalization, in turn, requires that affected interests recognize administrative legitimacy.

In addition to the new organizations and procedures, the administrability of the incentive system will be affected by the extent that interactions are required between organizations. Will federal, state, and local authorities be required to concur on all or some decisions? How will the various layers of decision makers coordinate their activities? Will decisions be made on a hierarchical or a tiered basis? What will be the relationship of community residents to the community and to the facility operators in implementing the incentive? What will be the relationship of the community, community residents, and facility operators to those responsible for implementing the incentive? (See subsection on Control and Authority.)

The ease of administration will also be affected by the complexity of potential appellate and renegotiation procedures that may accompany the incentive system. Although one objective of the incentive system is to establish confidence in the responsibility and liability of the facility operator to the community and its residents, occasions may arise when the established procedures and associated outcomes or impacts are inappropriate because of unanticipated events and/or trends. Since these situations may adversely affect either the community and some or all of its residents or the facility operator, procedures for resolving appeals and negotiations may be needed. In turn, the complexity of these procedures can affect how easily the incentive system can be administered.

Prior determination of the suitability and adequacy of an incentive system for a particular facility depends on accurate forecasting of project-related and incentive-related impacts. The projected impacts, in turn, affect the abilities of recipient individuals and jurisdictions to plan and to budget. If the incentive system does not encourage accurate forecasting of project-related and incentive-related costs and benefits, incentive recipients will not be able to plan or budget with any confidence. Relevant disciplines, particularly risk assessment, are immature and may seriously constrain accurate forecasting by recipient individuals and jurisdictions and, thus, complicate the administration of the incentive system.

Finally, the ease of incentive administration may be jeopardized by disproportionate administrative costs, which will vary according to the complexity of the administrative system, the number of governmental units involved in it, the complexity of the appellate-renegotiation process, and the sophistication and accuracy of impact forecasting methodologies.

Each of these five factors or dimensions can significantly alter the ease and costs of incentive administration. A simple, inexpensive administrative system is preferred unless it negatively affects the satisfaction of other important criteria. For example, if community perceptions of the fairness of incentive distribution (i.e., equity) and the legitimacy of the distribution system would be impaired by the simplest and least expensive appellate-renegotiation procedure (i.e., no procedure at all), this would not be an optimal solution; it would decrease or eliminate the possibility of siting the facility in or near the community.

Criteria Characterizing Community Understanding and Perception. A comprehensive evaluation of the utility of any incentive system requires an assessment of the degree to which the proposed incentive fulfills its role of mollifying public opposition. The preceding section set out a number of criteria for characterizing the incentive itself; this section discusses criteria for characterizing how members of the local community perceive the incentive system presented to them. As Kasperson (1980) noted, "One of the troublesome issues in the current efforts to formulate an acceptable solution to nuclear waste management is that our understanding of the problem *as it is defined by the public* continues to lag seriously behind our technical and managerial accomplishments" (italics ours). How the incentive system is translated by the community, then, is one of the fundamental determinants of whether the incentive will actually succeed in overcoming community opposition.

Although local community perception of the incentive system could probably be characterized by any number of criteria, we consider the three most important: (1) Is the community aware of the incentive? (2) Do community residents understand it? (3) Do they feel it is relevant to their concerns?

Local Knowledge. A necessary first step is to determine the extent to which the community is aware of the existence of an incentive. Members of the potential host community will probably be aware of the proposed waste repository. National publicity is given environmental problems, particularly those involving radioactive and hazardous wastes after the incidents at Three Mile Island and Love Canal (Duberg et al., 1980), and local publicity will be given to a particular facility. This knowledge of the facility itself, however, may tend to overshadow awareness of a proposed incentive or any influence the incentive may have in shaping public support or opposition. Of course, if the local public is unaware of the incentive, it obviously will not influence public support or opposition.

Interpretability. Once the extent of local awareness has been established, the next question is, How well does the local community understand what is being offered? This understanding is related to both the structure and the function of the incentive system.

An understanding of the structure refers to the ability of the community to define how the incentive will be implemented and will operate. This, in turn, requires the community to interpret many of the parameters of the incentive, e.g., certainty, constancy, and adequacy. How these parameters are interpreted will be based, in part, on the community's awareness of the siting process, past experience, and media coverage.

The function of the incentive system refers to the purpose the incentive is to serve. Interpretation in this regard has to do with the extent to which the community can understand the reason for being offered an incentive. For example, some members of the community might interpret any incentive as an indication that the facility is a greater risk than developers had led them to believe. Questions may be raised: "If this facility is so safe, why are they paying us to take it?"

Alternatively, some groups within the community may be unable or unwilling to distinguish the purposes of various legitimate incentives. This response may be less a reaction to *what* is being proposed than to *how* it is proposed. For example, an incentive may be less acceptable when the proposal is initiated by the facility sponsor than when it is developed in response to a community request (Rankin, 1981). Until incentive systems become much more common and accepted in siting, however, this response is likely to remain fairly typical.

In the case of both the structure and function of an incentive system, the more the public understands its actual goals and the true purpose for which it is proposed, the more likely the incentive system is to be able to contribute to the successful siting of the proposed facility.

Relevance. After the extent of community awareness and the level of understanding of the purpose and operation of the incentive have been determined, the next question is, To what degree is the incentive perceived to address or relate to the risks and impacts believed by the community residents to be associated with the facility? This assessment is critical in determining community acceptance of the incentive. If it is viewed as correcting problems associated with a facility, the incentive may be more favorably received by the public.

This criterion is closely related to that of interpretability, and, indeed, one of the major problems in assessing it is separating a lack of understanding from a lack of belief that the incentive will be implemented or is proper. Further, this criterion may generate intergroup conflict within the community; various interest groups may have different perspectives on what is a relevant incentive. Although this fragmentation may seem an insurmountable problem, Lindblom (1965) suggested that it may actually develop more innovative arrangements and encourage the participation of a broad range of interest groups in the implementation process.

Measuring all three of the criteria related to community characteristics may require a survey of public awareness of the incentive, with additional questions to probe the breadth and depth of understanding. A poll of opinions about the incentive's relevance might also be in order. Other appropriate information-gathering techniques might include a content analysis of relevant public meetings and local news media—particularly related stories, editorials, and letters to the editor—and discussions with local leaders.

Criteria Characterizing Consequences. The consequences of implementing an incentive can be analyzed by assessing its distributional effects and its propensity to affect community conflict or consensus. These effects are partially a function of the other incentive criteria (i.e., preconditions, an incentive's objective features, and community understanding of the incentive) and partially a function of the community's existing demographic, cultural, normative, social, political, and economic structure. Since much information in these areas has already been presented, the following sections summarize the major points of discussion.

Distributional Effects. Distribution refers to how the benefits, risks, and costs of a waste facility are received or borne by different individuals and/or groups in the community and beyond. In simple terms: Who benefits? Who pays? How do these effects accrue over time? Although the definition of fair distribution differs from person to person, it is generally agreed that beneficiaries of actions should pay the accompanying costs to the extent possible. The beneficiaries of the activities that have generated high-level commercial radioactive waste can be considered to be national, because nuclear power contributes to energy independence. A narrower view claims a smaller class of beneficiaries—the consumers of the electricity generated by nuclear power.

In dealing with equity or distributional effects, the states have clearly indicated that where physically possible, waste should be stored in the state where it originates (Kevin, 1980). This is not always feasible; many states lack suitable geologic media, and the cost of siting more than a few radioactive waste repositories may be prohibitive. The concentration of nuclear power use along the eastern seaboard and the Great Lakes and the concentration of suitable sites for nuclear waste disposal in remote, sparsely populated or arid areas of the United States, primarily in the west (Ausness, 1979), create the possibility of a serious spatial maldistribution of benefits and burdens.

The distributional effects of alternative incentive schemes (i.e., mitigation, compensation, and reward) are likely to vary. Mitigation mechanisms would be designed to prevent some impacts and reimburse for any unavoidable costs associated with the construction and normal operation of the facility. The impacts of mitigation are likely to be bimodally distributed on a geographic basis. Persons adjacent to a facility would be allocated a disproportionate share of some mitigation mechanisms, as measured on a per capita basis. Ideally, however, their share of mitigation would be proportional to the actual costs and risks they would be expected to bear. Other mitigation measures, such as public education and monies to reimburse communities for local capital expenditures to meet the demands of facility construction and operations, would be allocated to all citizens through their local governmental jurisdiction. The temporal distribution of mitigation could be designed in a variety of ways to accommodate current and future generations.

Compensation would be distributed to individuals who actually suffer harm as a consequence of an accident or other operational anomaly. The compensation system could be designed to ensure a fair distribution of resources. The actual allocation of compensation might be skewed, however, as a consequence of local inequalities in terms of access to compensation institutions (e.g., the courts in the case of insurance schemes). Efforts could be taken to assure equal access to compensation through the establishment and maintenance of sufficient legal resources to protect the interests of the poor, those normally unable to use the judicial system fully in their own interest.

It might be particularly difficult to design an incentive system that allocates rewards fairly. Unless substantial care is taken in designing and implementing the incentives, the rewards identified in the Rewards section of this paper (i.e., direct payments, bonus community services, tax incentives, advance payments, infrastructure development, and avoidance of other noxious facilities) could accrue to members of the local community equally, with no special provision being made for persons subject to greater impact (i.e., those directly adjacent to the facility). Although equal distribution may be appropriate for rewarding some of the costs and risks assumed by the local community (e.g., change in community character and the stigma of being a dumping ground), other rewards may need to be differentially allocated according to residential proximity to the facility and/or work at the facility. In these cases some of the rewards mentioned might be modified to acknowledge special situations; particular governmental services and programs could be targeted in these areas. Similarly, reward systems could be designed so that future generations are rewarded for decisions made by their forebearers; it might be required, for instance, that this part of the incentive be renegotiated periodically.

Generation of Local Conflict or Consensus. The siting of any noxious facility, such as a radioactive waste repository, can be expected to generate local conflict and opposition (Environmental Protection Agency, 1979; O'Hare, 1977). Presumably incentives will help diffuse some of that opposition, develop a local consensus supporting the siting and operation of the facility, and maintain that consensus within the host community for a period of 50, 100, or 500 yr.

Some incentive systems may themselves generate additional conflict and opposition within the potential host community, however (i.e., conflict over and above that generated by the siting of the facility). Although it may be difficult to distinguish empirically between the causes of conflict, particularly because of their potential interrelatedness, it is essential to distinguish between them conceptually.

To the extent that the two interventions (i.e., facility and incentive) are perceived by community members as a single issue, conflict or opposition to the incentive would be expected to spill over and affect the siting decision. Conflict generated by the incentive may polarize the community and jeopardize the integration of the facility and its operators into the community if the facility is indeed sited there. If local community residents perceive that the incentive is offered to divert their concern and attention away from facility impacts, the incentive may exacerbate local opposition. This perception would likely undermine whatever confidence local residents and officials have in the reliability and intentions of the federal government, the state government, and the licensing and regulatory procedures—in short, their confidence and trust in external authorities and decision makers.

The possibility of conflict over the facility and the incentive should not obscure the significant role an incentive system can have in developing community consensus supporting the siting of a repository. As indicated in the section entitled Incentives: Some Empirical Data, there is some evidence that incentives increase local support for hosting a repository and that that support is predicated on a multidimensional incentive system—one that includes, for instance, transfer payments, access to information, and participation in repository operation. Any of these factors alone is insufficient to the development of consensus.

It is important to recognize that consensus, as an operational concept, is in itself multidimensional. From a quantitative standpoint, it is not certain how much support is required before a consensus is attained—certainly more than a majority and perhaps less than unanimity. Conventionally approximately two-thirds to three-fourths support of a particular action by a given population can be considered community consensus. A consensus is required rather than a simple majority to minimize the possibility of losing so much support over time that the initial decision would be reversed. In a policy arena such as radioactive waste repository siting, where a reversal could have significant adverse impacts (e.g., litigation, if not outright closure), achieving a substantial consensus supporting the facility is obviously important.

The negotiations involved in siting and an incentive design process can be a significant impetus toward the development of community consensus. What is critically important, as mentioned previously, is creating the opportunity for all interested parties to be represented in negotiations. In this way divergent attitudes and reasons for those attitudes can be discovered, and the community can determine what package of benefits would be necessary to develop consensus. Negotiations will not automatically produce consensus, but they are likely to increase substantially the possibility of consensus.

The determinants of the conflict-consensus criterion are quite varied. They include site- and community-specific features (e.g., political culture, degree of urbanization, population heterogeneity, and socioeconomic status), input factors (i.e., project- and incentive-related inputs), and contextual variables (e.g., trust, method of presentation, and local participation). With respect to incentive-related inputs, they include factors such as the interpretability of the incentive system, the extent of local control in the development of the incentive system, who pays for the incentive, and the relevance and salience of the incentive system.

Resolution of Adverse Consequences. Given the stated purpose of incentives in radioactive waste repository siting—to achieve a balance of real and perceived local burdens and benefits to encourage local acceptance of the facility—the incentives themselves may seem to have adverse consequences. Such a possibility is real, and we cannot even assure the reader that we have identified all such consequences. What can policymakers do to minimize the likelihood of such occurrences?

Awareness of the possibility is the first step. More important, however, we can systematically evaluate the causes of such outcomes and design a siting process that is sensitive to these phenomena. The activities can be approached in a variety of ways, but, given the centrality of local community acceptance of the repository, it is essential that siting and incentives be assessed in a community context. The framework and criteria described in this chapter, though based on a comprehensive review of relevant literature, were developed to a large extent inductively. Their actual utility to potential siting decisions can be judged effectively only by members of a potential host community.

Encouraging community self-examination of the risks, costs, and benefits of repositories and associated incentives may be advantageous to the federal government. Independent reviews and assessments have been suggested by various interests (Painter, 1981; National Governors Association, 1981), and potential models are available (National Conference of State Legislatures, 1980; Carnes, et al., 1982). Experimenting with community participation in repository siting, incentive design, and implementation processes not only would help to corroborate or reject totally or in part the analyses offered here but also would provide one of the most relevant ways in which to evaluate costs, risks, and benefits.

Incentives: Some Empirical Data

Results of a survey of 420 Wisconsin residents in three communities* give us some evidence about incentives. Although the applicability of this kind of evidence is limited, as discussed below, the results of this survey provide some interesting illustrations of what people say they feel about radioactive waste disposal and potential incentives that might alter their positions and attitudes.

When asked initially if they favored or opposed the construction of a nuclear waste repository in their community, 6% strongly favored such a project, 16% mildly favored it, 7% mildly opposed it, and 64% strongly opposed it. Seven percent replied that they didn't know or failed to respond. Later in the survey, respondents were presented with a list of incentives and asked if the provision of a particular incentive or package of incentives would represent a sufficient level of local control over a nuclear waste repository. The phrase "a sufficient level of local control" was carefully chosen. The incentives presented to the respondents are listed in Table 1.

Although the survey was conducted before the development of the conceptual framework described in this paper, the incentives used in the survey fit into the framework. Access to information and monitoring are clearly mitigation incentives. Substantial payments to the community were intended to be compensation. Representation and the power to shut down the facility are less clear but might be thought of as preconditions.

The incentives were ordered to reflect what the researchers felt was an increasing level of community control. Each new incentive was added to the previous one until the respondent indicated that the package of incentives would be sufficient.

Poll conducted by John Kelly, Complex Systems Group, University of New Hampshire and data reconstructed by John H. Reed, Oak Ridge National Laboratory, for this application.

Table 1 Distribution of the Acceptance of Incentives to Site a Nuclear Waste Repository Among Wisconsin Residents

Incentive	Percent
Substantial payments to your community	17
Access to information*	13
Independent monitoring*	12
Representation on a governing board of the facility*	10
Power to shut the facility down*	11
Other*	3
Unacceptable under any condition	17
Do not know—no answer	16
Total	99
Number sampled, 426	

**These figures represent the persons who would have chosen the particular incentive and all those preceding it on the list.*

What is remarkable about this distribution is its relative uniformity. The two largest categories (17% each) include those who would accept substantial payments and those who would find a repository unacceptable under any circumstances. The addition of incentives of information, monitoring, representation, or the power to shut down the repository each added about 10% of the respondents. At face value this suggests that each of these incentives would encourage more people to accept a waste repository.

Asking if people perceive that a package of incentives would give greater control is a different matter from determining whether people would change their opinions if they were given a package of incentives. The incentive question was followed by one that asked whether the respondent would accept a waste repository in the community if the incentive package he/she chose was provided. Comparing this response to an earlier question about a repository in the community gave a direct measure of whether people say they would change their opinions. The percentage of those who said they would accept a repository increased from about 22% without incentives to about 42% when incentives were available.

Table 2 shows the distribution of persons who said they would change their opinion to accept a repository in their community if given a package of incentives. Having access to information had the most impact. This was followed by payments to the commu-

Table 2 Distribution of Persons Who Changed Their Position to Accept a Waste Repository in Their Community by Type of Incentive Package*

Incentive	Cumulative percent
Substantial payments to your community	23
Access to information	48
Independent monitoring	65
Representation on a governing board of the facility	74
Power to shut the facility down	93
Other	99
Total, 77 persons	

**Incentives were offered additively, i.e., people were asked if substantial payments to the community would be sufficient to accept a repository. If they replied in the negative, they were then asked if payments* and *information would be sufficient. The interviewer continued down the list until there was an affirmative response or until the list was exhausted.*

nity, the power to shut down the facility, independent monitoring, and representation on a governing board. An important observation to be made from this distribution is that there is no one incentive that leads a majority of people to change their opinions.

Obviously, this limited survey does not provide a definitive analysis of incentives. It does suggest several things, however. First, incentives can encourage people to change their positions about accepting the siting of waste repositories. The level of acceptance of incentives and of waste repositories would change with specific conditions in local communities and with the nature of the negotiations over incentives. Second, the data suggest that noneconomic incentives may contribute significantly to public acceptance of a radioactive waste repository in a local community. Third, the data suggest that packages of incentives rather than single incentives may be required to gain acceptance. Finally, this very limited exploratory approach suggests that survey methodology can be used to explore people's responses to incentives.

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Economic and Fiscal Impacts of Large-Scale Development Projects: Implications for Nuclear Waste Repositories

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This paper deals with the local economic and fiscal implications of siting high-level nuclear waste repositories in rural areas. The economic and fiscal effects of repository development fall into two categories:

1. Standard impacts similar to those that would be associated with developing any large-scale industrial facility in an isolated area.

2. Special impacts that result from the hazardous nature of the nuclear materials stored and from federal ownership of the facility.

Standard economic and fiscal impacts include employment effects (direct and secondary), local income changes, alterations in community price structures, effects on community services, and changes in revenues and costs for local jurisdictions. Special impacts include the possibility of diminished activity in other basic economic sectors, negative effects on the area's long-term growth prospects and a consequent dampening of investment in the local trade and service sectors, additional costs for local jurisdictions (e.g., for preparing evacuation plans), and limited local tax revenues resulting from the tax-exempt status of the facility. These special effects are difficult to quantify and require additional analysis. 47 ref.
1 fig, 4 tab.

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Preliminary identification of alternative sites for nuclear waste repositories indicates that many of these sites may be located in rural areas of the nation (U. S. Department of Energy, 1980). Construction, operation, and maintenance of such facilities will lead to significant socioeconomic impacts for these rural areas. These impacts will result from the waste repositories being large-scale developments located within a rural context (these impacts can be referred to as standard development impacts) and from the special socioeconomic effects resulting from the hazardous nature of the nuclear materials stored at repository sites (these impacts are termed special effects or impacts). If rural areas are to be equitably compensated for the national service they will perform in housing such repositories, investigators must identify the impacts of these repositories; and mitigation, community development, and public participation plans must be developed on the basis of a careful program of research and local area involvement.

The Socioeconomic Analysis of Nuclear Repository Siting project addresses the economic, demographic, public service, fiscal, and social impacts associated with repository siting, including both standard and special effects. Objectives of this project are:

1. To develop methodologies for assessing the socioeconomic impacts of repository siting
2. To examine the broad regional and local economic implications of such siting
3. To develop impact mitigation, community development, and public participation strategies that can be applied in rural siting areas across the country

The purpose of this paper is to summarize the major activities of this project relating to the economic and fiscal implications of repository siting.

Results and Implications of Recent Research and Development Activities

Recent key activities of the project related to economic and fiscal impacts were:

1. Development of a generic description of the standard and special effects of repository siting
2. Completion of a detailed review of economic and fiscal impact assessment techniques, and the formation of preliminary recommendations for the development of economic and fiscal impact methodologies for use in repository siting
3. Development of a plan of work for future years

Each of these activity areas is discussed briefly below.

Standard and Special Impacts—Economic and Fiscal.

Impacts of repository projects on rural areas result in part from such projects being large-scale activities relative to the areas where they will be located. These facilities will involve 1400 to 3100 workers during construction, 900 to 1200 workers during operation, and the facilities probably will require monitoring for an indefinite period (U. S. Department of Energy, 1979). Although actual sites have not been selected, many of the most geologically acceptable prototype sites, particularly in the West and Southwest, are in relatively sparsely settled rural areas that have unique social and cultural conditions (U. S. Department of Energy, 1980). For such prototype sites as those in Anderson, Freestone, and Leon Counties in Texas (1980 populations of 38,381, 14,830, and 9594, respectively); Bienville Parish, Louisiana (1980 population, 16,387); Perry County, Mississippi (1980 population, 9864); and San Juan County, Utah (1980 population, 12,253) (U. S. Bureau of the Census, 1981), the relative magnitude of the impacts of such a development would be significant. Thus siting, construction, and operation of nuclear waste repositories may involve factors, and probably will have many impacts, that are similar to those for other large development projects in rural areas.

On the other hand, repositories also will have effects that are unique or special because they are nuclear repository sites and are subject to the effects that result from public perceptions of nuclear power and radioactive wastes. These special impacts are difficult to identify and require extensive analysis; nevertheless both standard and special effects must be identified if mitigation efforts are to be successful. Some of the most important standard and special effects of repository projects are identified below, and a brief discussion of the factors determining the magnitude and distribution of such impacts is presented.

Factors Affecting the Socioeconomic Impacts of Large-Scale Projects. The magnitude and distribution of the standard socioeconomic impacts of large-scale projects can be seen as a function of the following two general sets of factors:

1. Characteristics of the project
2. Characteristics of the site area (Murdock and Leistritz, 1979).

The unique combination of these factors and the interaction of these characteristics determine the nature and the magnitude of socioeconomic impacts in rural areas.

The characteristics of a project play a central role in the determination of impacts. In particular, a project's location, its level and type of direct employment requirements, its potential to produce secondary (indirect and induced) employment, the length of its construction and operational phases, and the employment policies of the developer all may affect the level of impacts. For example, if a project is located near a relatively large population center, its direct employment requirements are small, its potential for generating secondary employment is limited, its construction period is relatively long, and its developer employs a large number of existing residents, then the population growth in the impact area and the socioeconomic impacts of that growth will be relatively small. On the other hand, if the project is located in an area distant from any major population center, has large direct and secondary employment requirements, has an expedited construction schedule, and has few local residents employed by the developer, then in-migration to the site area will be extensive and the impacts will be significant (Leistritz and Murdock, 1981).

The socioeconomic characteristics of the site area and its residents also will affect the magnitude and distribution of socioeconomic impacts. Such factors as the number of alternative settlement sites in the impact area, the skill levels of the local labor force, the availability of local labor for employment at the project, the level of development of local community service and organizational structures, and the preferences of residents in local communities are among the most important of such factors. If there are a large number of alternative settlement sites, the new population is less likely to concentrate at any one site. The compatibility and availability of the labor skills of existing residents with those required by the project will determine how many jobs can be taken by indigenous residents and how many new residents must in-migrate. Existing service bases of communities and the growth preferences of community residents often serve to decrease or accentuate the attractiveness of communities to new residents and to alter both the number of new residents received by a community and the positive and/or negative impacts associated with that population growth. Because the areas tentatively identified as potential sites for repositories differ substantially with regard to the characteristics previously discussed, investigators will have to develop detailed site-specific assessments to estimate the impacts likely to be associated with developing a repository in each of these locales.

Characteristics of the project and the site area seldom operate independently, and it is the interaction of these two factors that determines the actual level of socioeconomic impacts. The complementary or countervailing nature of these factors can work to reduce or accentuate the socioeconomic impacts likely to result from any given project; consequently the interactive as well as the individual effects of these factors must be examined.

Identification of Standard Impacts—Economic and Fiscal. Standard economic and fiscal impacts of nuclear repository siting are defined as those effects resulting from such facilities being large-scale projects relative to the areas where they will be located. Requirements of the work force for these facilities are similar to those for large energy resource development facilities (see Table 1); and, if nuclear repositories are sited in sparsely settled rural areas, their effects on the local economy and on the fiscal status of local governments can be expected to be similar to the effects caused by large energy resource development facilities. Some of the major standard effects of repository siting that can be expected include changes in:

1. Employment patterns and characteristics
2. Income levels and distribution
3. Community price structures
4. Community service requirements and costs
5. Local government revenues and financing

These effects are discussed briefly below. For more detailed discussions, see Murdock and Leistritz (1979) and Leistritz and Murdock (1981).

Employment Effects. Requirements of the labor force for various types of energy facilities will be examined in this section. First, the total size and skill levels of the required work forces will be examined. Then the characteristics of workers at new energy facilities will be briefly summarized. Finally, the magnitude and timing of secondary employment effects will be examined.

Direct employment effects of a new energy facility or other large development project typically occur in two phases—the construction of the facility and its subsequent operation. Construction work forces for most types of facilities (except mines) are several times larger than the operational work force; however, the construction work forces are present in an impact area for only a few years, whereas operational work forces, which may be quite small, will be required for the life of the facility.

Although the employment levels required by specific projects may vary widely, investigators can discern some general patterns by examining typical sites. Examples of the size of work forces for several different types of energy facilities are presented in Table 1. Data in Table 1 indicate that large construction work forces can be expected for electric generating, synfuel, and oil shale plants as well as for nuclear waste repositories. In addition, synthetic gasification plants, oil shale processing plants, and nuclear repositories may have a thousand or more permanent operating employees. Clearly, the size of the work forces associated with such facilities is likely to have a relatively large effect on small rural communities.

An examination of the characteristics of energy-related work forces in several Northern Great Plains states indicates that substantial contrasts exist between construction work forces and permanent operating and maintenance work forces (Murdock and Leistritz, 1979). Construction work forces are dominated by craftsmen who have highly specialized skills and are geographically mobile in response to new job opportunities. Wages are high, employment is temporary (average job tenure is about two years), and some job opportunities are created for local workers. For example, a 1975 survey of workers at 14 construction sites indicated that local workers made up about 40% of the total construction work force. Local workers, most prevalent in the less skilled job categories, occupied more than half of the unskilled labor

Table 1 Energy Facility Employment Characteristics*

Type of facility	Size	Construction period (years)	Construction work force (number at peak)	Operating work force (number)
Surface coal mine†	9,000,000 tons/year	2 to 3	150 to 200	325 to 475
Underground coal mine†	2,000,000 tons/year	2 to 3	225 to 325	550 to 830
Electric generating plant† (includes surface coal mine)	700 megawatts	4 to 6	750 to 1,050	90 to 170
	2,250 megawatts	6 to 8	2,000 to 3,000	350 to 650
Synthetic gasification plant† (includes surface coal mine)	250,000,000 cu ft/day	3 to 4	3,000 to 4,000	1,050 to 1,300
Oil shale processing facility† (includes mining)	50,000 barrels/day	3 to 4	2,000 to 2,400	1,050 to 1,450
Uranium mining and milling†	1,000 tons U_3O_8 concentrate/day	1 to 3	130 to 250	200 to 250
Nuclear waste repository:				
in salt‡	Reference site	3 to 5	1,800	870
in basalt‡	Reference site	3 to 5	3,100	1,100

**Employment figures are estimates for general planning purposes only. Actual employment for any particular facility will depend on size, type of equipment used, mining conditions, construction schedule, and numerous other factors.*

†Murdock and Leistritz, 1979.

‡U. S. Department of Energy, 1979.

positions (Mountain West Research, Inc., 1975).

Power plant and mine operations require substantial numbers of heavy equipment operators, mechanics, and craftsmen/technicians. Wages paid generally are higher than those prevalent in other jobs in rural areas (Wieland, Leistritz, and Murdock, 1977), and most firms appear to emphasize local hiring in order to build a stable work force. For example, a survey of workers at 14 power plants and coal mines in Montana, North Dakota, South Dakota, and Wyoming indicated that local workers made up 62% of these work forces, and only two sites had less than 50% local workers (Wieland, Leistritz, and Murdock, 1977). On-the-job training and internal promotion are used to fill skilled positions; however, a small cadre of experienced workers may be transferred from other plants or mines when a new project begins operation. Nonlocal construction and permanent workers most frequently come from within the state or from adjacent states.

Hiring practices of energy firms emphasize employing workers with some experience in equipment operation and maintenance. Hence the energy industry draws heavily from local trade and service firms and particularly from the local construction industry (Leholm, Leistritz, and Murdock, 1975; Dobbs and Kiner, 1974). Thus substantial numbers of local workers may be able to participate in the new job opportunities created by energy development; however, local construction companies, trade and service firms, agricultural producers, and local government units may face severe competition for labor if extensive development occurs (Baker, 1977). These employers may find it necessary to increase their rates of compensation or face the prospect of losing their most productive workers.

In addition to the employment created directly in facility construction and plant and mine operation, energy development stimulates economic activity and employment in various trade and service sectors of the local economy. The additional employment created indirectly as the result of a project often is termed secondary employment, service employment, or indirect and induced employment. Questions surrounding the secondary economic impacts of energy development include the magnitude of local secondary employment and income effects, the timing of these effects in relation to the project schedule, the location of secondary economic effects, and the occupational composition of employment effects. Unfortunately, although many analysts have attempted to project the secondary employment effects of energy development, data regarding actual employment responses to major energy projects are very limited.

A summary of employment multipliers used in estimating secondary employment effects of western energy development projects is shown in Table 2. All estimates shown are based on projected employment changes. In most cases the employment multipliers are smaller during the project construction phase than during the subsequent operation phase. Construction-phase multipliers (change in total employment/change in direct employment) range from 1.2 to 2.1, whereas operation-phase multipliers range from 1.7 to 2.5. Multipliers shown in Table 2 generally are regional multipliers reflecting additional indirect and induced employment created in a multicounty trade area (also sometimes called a functional economic area). Thus only a portion of the secondary employment resulting from a project would occur in the site county; the remainder would occur in trade and service

Table 2 Estimated Secondary Employment Associated with Western Energy Development Projects

Project type	Project location (county, state)	Project phase	Direct employment	Secondary employment	Employment multiplier (total employment/ direct employment)
Electric generating/coal gasification complex with surface mine	Mercer, North Dakota	Construction	3240	1004	1.31
		Operation	1314	1449	2.10
Coal surface mine	McLean, North Dakota	Construction	270	139	1.51
		Operation	225	194	1.86
Oil shale mining and refining	Rio Blanco, Colorado	Construction	1200	450	1.38
		Operation	1050	975	1.93
Coal-oil-gas complex and coal surface mine	Slope, North Dakota	Construction	5050	1010	1.20
		Operation	2630	1972	1.75
Coal-oil-gas complex and coal surface mine	Oliver, North Dakota	Construction	5050	2525	1.50
		Operation	2630	3682	2.40
Uranium mine and mill	Converse, Wyoming	Construction	230	253	2.10
		Operation	220	330	2.50
Coal surface mining (multiple mines)	Campbell County, Wyoming	Construction	2727	1013	1.37
		Operation	2788	2034	1.73
Electric generating plant with coal surface mine	McLean County, North Dakota	Construction	1270	622	1.49
		Operation	485	622	2.28
Electric generating plant with coal surface mine	Grimes County, Texas	Construction	550	385	1.70
		Operation	250	175	1.70

Sources: Toman et al. (1978); Colony Development Operation (1975); Gilmore et al. (1975); Jackson (1977); Hayen and Watts (1975); Toman et al. (1976); and Resource Communities, Inc. (1977).

centers in nearby counties.* This pattern is likely particularly during the construction phase when a sizeable portion of the construction work force may be commuting substantial distances to the project site.

One recent study included a retrospective analysis of secondary employment effects. Gilmore and his associates (1981) conducted case studies of the socioeconomic effects of power plant construction at several sites across the country. Their estimates of the secondary employment effects of nine of these projects are presented in Table 3. Examination of this table indicates that the magnitude of the secondary employment effects at these sites was substantially smaller than had been assumed in many anticipatory impact assessments; however, the expected overall patterns were exhibited (see Table 2). Thus secondary employment multipliers were greater during the operational period of the project than during the construction phase; there also was a general tendency for the multipliers to be smaller for more sparsely populated areas (e.g., Boardman, Coronado, Laramie River) than for areas with larger population and economic bases (e.g., Bellefonte, Clay Boswell).

On the basis of these case study results, Gilmore and his associates (1981) estimated the relation between site area characteristics and the magnitude of secondary employment effects. These estimates are presented in Table 4. As indicated in this table, sparsely populated areas with poorly developed trade and service sectors can be expected to have smaller multipliers than areas with more diversified economies.†

Although the timing of secondary employment effects is an important consideration in economic and social impact assessment, little empirical evidence is available on this impact dimension. Many analysts have assumed lag effects in secondary employment responses, but few have attempted to quantify these effects. A recent study by Conopask (1978) includes an explicit effort to measure these lag effects. The study used a pooling of cross-sectional data for 15 Northern Great Plains coal development counties over a five-year period. Multiplier effects of mining, manufacturing, and construction employment were estimated using regression analysis. Only mining employment was found to have a statistically significant lag effect in the best-fit equation. Construction employment was found to have a significant lag effect in some specifications of the model, but not in others. This study together with the study by Thompson, Blevins, and Watts (1978), which indicated only small lag effects, and the study by Gilmore and Duff (1975), which indicated large lag effects, suggest that more analysis of this topic is essential.

Information on the occupation and industrial composition of secondary employment is even more limited. Occupations of heads of households who had moved to energy-impacted communities since the start of energy construction activities are reported by Mountain West Research, Inc. (1975). Among household heads who were not directly employed in energy construction activities, the most numerous occupational groups were professional, managerial, technical, and kindred workers; craftsmen, foremen, and kindred workers; and laborers.

**For attempts to estimate these effects using models that combine export base and central place concepts, see Chalmers et al. (1977) and Bender (1975).*

† Recent work by Temple (1978), on the other hand, indicates that a given change in basic activity generates a larger change in nonbasic activity in areas with smaller economic bases. This effect may be attributable to scale effects or to the existence of excess capacity in areas with larger economic bases.

Table 3 Estimates of the Ratio of Local Service to Construction and Operating Employment, Selected Power Plant Construction Sites*†

	Ratio of local service employment to construction workers residing in the impact area		Ratio of local service employment to operating workers‡ residing in the impact area	
Case study, state	**Peak construction employment**	**Annual average construction employment**	**Annual average operating employment**	**Annual average in-migrating operating employment**
Coal Creek, North Dakota	0.4	0.4 to 0.5	0.8	0.8 to 0.9
Clay Boswell, Minnesota	0.6	0.6 to 0.7	0.7 to 0.8	0.8
Boardman, Oregon	0.1 to 0.2	0.2 to 0.3	0.2 to 0.3	0.3
Laramie River, Wyoming	0.1 to 0.2	0.2 to 0.3	0.6 to 0.8	0.8
Newton, Illinois	0.2	0.2 to 0.3	0.6 to 0.8	0.8
Fayette, Texas	0.2 to 0.3	0.3 to 0.4	0.6	0.6 to 0.7
Bellefonte, Alabama	0.6	0.6 to 0.7	0.7	0.7 to 0.8
Coronado, Arizona	0.1 to 0.2	0.2 to 0.3	0.6	0.6 to 0.7
Cholla, Arizona	0.2 to 0.3	0.3 to 0.4	0.6	0.6 to 0.7

**Gilmore et al., 1981.*

†Estimates presented in this table are of the form (E_s/E_b), where E_s = change in secondary employment and E_b = change in basic employment (e.g., construction or operations workers). Add 1.0 to each estimate to make these estimates comparable to the employment multipliers presented in Table 2.

‡As a rule, estimates of the operating work force multipliers were made during the buildup of the operating work force; multipliers probably will be higher under steady-state conditions.

Table 4 Estimates of the Relation Between Site Area (Impact Area) Characteristics and Secondary Employment Ratios*†

Type of site area	Ratio of secondary employment to in-migrating basic workers: In-migrating construction workers: Ratio to peak	Ratio of secondary employment to in-migrating basic workers: In-migrating construction workers: Ratio to annual average	Ratio of secondary employment to in-migrating basic workers: In-migrating operating employees
Rural, sparsely populated, no large trade center within impact area	.1 to .2	.2 to .3	.3 to .5
More urbanized impact areas, population densities moderate	.1 to .2	.3 to .4	.4 to .6
On the fringe of a metropolitan area	.2 to .3	.4 to .5	.6 to .8

**Gilmore et al., 1981.*

†As in Table 3, estimates are of the form (E_s/E_b). Add 1.0 to each estimate to compare to employment multipliers presented in Table 2.

Hannah and Mosier (1977) report on findings of case studies in several energy-impacted areas in Montana, Wyoming, and Utah. Although no quantitative data on secondary employment requirements are presented, the report identifies a number of industry and occupational categories where personnel shortages frequently are experienced. These include non-energy construction (carpenters, electricians, plumbers, and brick masons), public agency (police, maintenance workers, and clerical workers), commercial service (retail sales clerks, food service workers, auto repair and maintenance workers, and motel/hotel maids), and health service (doctors, dentists, and technicians) occupations.

To summarize, evidence concerning the actual magnitude and timing of indirect employment associated with new energy facilities is limited, and information that does exist presents contradictory findings. Even less is known about the industrial and occupational composition of indirect employment as well as the geographical origins of indirect workers. These topics are priority areas for future research.

Income Effects. Development of a major project will result in substantial changes in income in the affected area. As with employment, both direct and secondary effects are important, and income multipliers frequently are used to estimate the magnitude of secondary income effects. Research aimed at quantifying the income effects of major resource development projects has, however, been even more limited than that for employment. Several reasons have contributed to income effects of new projects receiving less attention than the employment effects. Income data are less readily available from secondary sources than employment data, and businesses often are reluctant to release wage and salary information (Denver Research Institute, 1979). Obtaining reliable estimates of nonwage income is frequently difficult, and the task of dividing nonwage income into basic and nonbasic components is even more difficult. Finally, employment frequently is used as a target variable by planners because changes in employment can be readily related to changes in population and public service requirements. As a result, employment effects have been given greater research emphasis than income effects.*

Direct income effects of a project depend primarily on the facility's payroll and on the extent to which it purchases supplies and materials locally. Average salaries and wages during the construction phase of a project typically are quite high relative to the wage and salary levels prevailing in rural areas. Leakages of purchasing power from the local area near the site also may be high during this phase, particularly if a large proportion of the construction workers commute from outside the area. Indirect income effects of a project depend on the same factors that determine indirect employment effects. Larger more diversified economies generally will have larger income multipliers.† For example, Tweeten and Brinkman (1976) report that income multipliers frequently are of a magnitude of 2.0 for multicounty districts; however, income multipliers for individual communities often are in the range of 1.0 to 1.5 (i.e., a multiplier of 1.0 indicates no additional net effect beyond the direct income increase). Chalmers et al. (1977) report similar findings from a study of counties in the Northern Great Plains. For regional trade center counties, which have populations in excess of 15,000 and personal income in their market areas in excess of $100 million (in 1977 dollars), the average income multiplier was 2.02. For second order counties having populations between 5000 and 15,000 and market area personal income exceeding $20 million per year, the average income multiplier was 1.66. For the remaining counties, which have populations less than 5000, the average multiplier was 1.23.

Gordon and Mulkey (1978) report results similar to those noted above. On the basis of a review of community input-output studies, they conclude that income multipliers for small communities are generally in the range of 1.1 to 1.5, whereas larger communities (or functional economic areas) may have multipliers of 1.5 to 2.5. These authors point out, however, that these multiplier values represent an average for the entire local economy and that larger multiplier values may be associated with growth in certain local sectors.

A major question with respect to the income effects of a project relates to the distribution of these effects among the local population. Although it appears that local workers who obtain employment at new energy projects experience substantial increases in their incomes (Murdock and Leistritz, 1979) and that the effect of most projects will be to increase average per capita income levels in the site area, investigators know little about the effects of these projects on income distribution. A recent study addresses the effects of industrial development on incomes and the income distribution of rural industrial workers in Texas (Reinschmiedt and Jones, 1977). Data from nine industrial plants in six Texas communities with populations of less than 15,000 indicated that 80% of all workers increased or maintained their previous earnings when they took jobs at the plants. The analysis also indicated a slight increase in overall income dis-

**This tendency for planners to focus on employment effects and multipliers may not be entirely desirable. For example, some sectors may have substantial income effects, but only modest employment effects (Scott and Braschler, 1975). Both effects should be considered in development planning.*

†For a more detailed and technical explanation of employment and income multipliers, see Miernyk (1965), Schmid (1978), Lamphear and Emerson (1975), and Isard (1960).

tribution equality among these workers (Reinschmiedt and Jones, 1977). Individuals in the lowest income categories before taking jobs with the plant experienced the greatest income gains (partly because 27% of the workers previously had been unemployed).

In summary, little attention has been given to measuring the income effects of new resource developments or to discerning the distribution of these income effects.* These topics should receive extensive attention in future research efforts.

Impacts on Community Price Structures. One of the impacts that often affects energy development areas is an increase in prices resulting from increased demands for many goods and services. Housing costs particularly are susceptible to such effects and often have been found to increase rapidly during periods of energy-related growth (Gilmore and Duff, 1975). Such increases may create price structures that severely limit the range of housing available to many residents (Bronder, Carlisle, and Savage, 1977).

Increases in prices may pose problems for certain groups. The elderly, those on fixed incomes, and others that are not directly associated with the energy developments may face significantly increased costs, but receive few income benefits. These persons, then, may suffer as a result of development.† This effect may be of particular concern in small rural towns that traditionally have served as retirement villages for rural and urban residents because of their relatively low cost of living (Scott and Summers, 1972). Changes in price structures have received insufficient attention and clearly require more careful analysis.

Impact on Community Services. Rapid population growth in communities near development sites often leads to increased demands for all forms of public services. The magnitude of the population increase, the rapidity with which the increase occurs, the amount of lead time provided to community leaders for planning, and the state of existing local service infrastructures will determine the degree of difficulty communities face in meeting increased service needs. When substantial population growth occurs with little advance warning, communities can experience severe growth-management problems (Gilmore and Duff, 1975).

Some forms of services pose particularly severe problems for local officials because substantial costs are associated with providing higher levels of these services. A review of selected studies of the public service impacts of energy developments indicates that substantial capital costs typically are associated with expanding schools and water and sewer systems (Leholm, Leistritz, and Hertsgaard, 1976; Gilmore et al., 1976; Murphy, 1975). These studies indicate that unless a community has excess capacity in its public service infrastructure before initiation of the project, the community can expect to experience additional capital costs of

**A related topic, which has received even less attention, is the effect of resource development on the distribution of wealth. Although the potential for substantial increases in property values generally is recognized (Tweeten and Brinkman, 1976; Bradley, 1978), only a few studies have attempted to quantify these effects (Debes, 1973); and we are aware of no study that examines the distributional aspects of these changes.*

†For example, Clemente and Summers (1973) found that the construction of a large steel mill in rural Illinois had a negative effect on the relative income status of the elderly residents of the area. Data presented by Scott and Summers (1972), on the other hand, suggest older persons increased their relative advantage during a period of rapid industrial development. In light of these conflicting results, additional research on this subject clearly is required.

$3000 to $6000 for each member of the in-migrating population and additional operating and maintenance costs of $400 to $600 per capita (in 1975 dollars). Therefore costs of providing expanded levels of services may pose severe cash-flow problems for local governments.

Some types of services may pose special problems because existing levels of these services often are inadequate in rural areas. Medical and mental health services typically fall into this category. In addition, recreational and cultural facilities often are in short supply in rural areas and law enforcement, fire protection, and emergency services frequently are inadequate to meet even predevelopment needs (Murdock and Leistritz, 1979).

Impact on Fiscal Structures. Increased service demands and costs created by rapid population increases in energy development areas often cause severe fiscal problems for local governments. These problems lie primarily in the timing of revenue collection in relation to public service costs and in the distribution of costs and revenues between jurisdictions. The problems are more extensive the larger the population influx is relative to the indigenous population base and the larger the differences between the size of the construction and the operational work forces.

Local jurisdictions often experience severe cash-flow problems during the development of a new project because, although new service demands arise immediately during construction of the project, many of the revenues necessary to meet those costs are not available until the operation of the project begins (see Fig. 1). This condition arises because local governments tend to be dependent on property taxes, but construction populations are likely to live in temporary housing with low taxable values, because secondary development of taxable business property occurs relatively slowly, and because energy related taxes, such as severance taxes, typically do not become available until the plant becomes operational. As a result, the revenue-cost squeeze may become critical (Toman et al., 1977; Gilmore et al., 1976). Local governments must decide whether to invest in local service structures during the construction period (when demands are high) and be faced with excess capacity to support during operational periods, or to muddle through with severely impacted service bases during construction periods and build service structures to meet the lower level of demands expected during operational periods. Equally problematic, local governments must attempt to convince local citizens of the need to increase taxes to pay for services when the uncertainties concerning actual new operational populations, the degree of local dissatisfaction with services and government, and fears about tax increases all are high.

Problems associated with the distribution of tax revenues between jurisdictions may be equally severe. The facilities and resources that generate new revenues may be located in one county, whereas the impact-related populations are located in a different county or even in a different state. In such cases, the governmental entity receiving the new revenue benefits will experience a tax windfall while the area with the influx of new population will be faced with especially severe fiscal problems.

Even when no geographical distribution problems exist, a problem may result from the fact that many tax structures distribute a majority of resource-related revenues to state or county governments; however, the costs often are greatest at the municipal or school district levels. Thus, for many rural areas experiencing

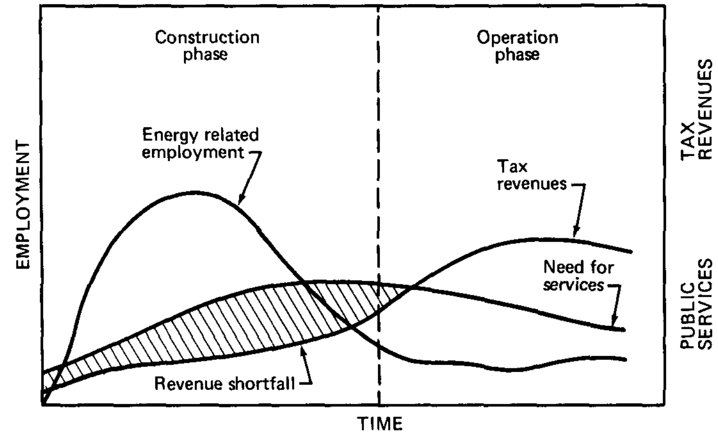

Figure 1 Energy related employment, tax revenues, and need for public services in an area affected by a large-scale energy development (Murdock and Leistritz, 1979).

impacts from large-scale projects, the major fiscal problems are related to the temporal and geographical distribution of revenue and not to the overall levels of revenue generated by the project (Gilmore et al., 1976; Stinson and Voelker, 1978; Gray et al., 1977; Toman et al., 1977).

In order to address these timing and distribution problems, investigators have suggested a number of impact assistance/mitigation programs (Gilmore et al., 1976; Auger et al., 1976; Leistritz and Murdock, 1981). These include federal and state loans and grants, tax prepayments by the developer to affected local areas to help them overcome revenue collection-timing problems, and the examination of alternative distribution formulas and local sales and income taxes to address fiscal locational problems. In any case, fiscal problems remain a major area of concern in the analysis of development impacts.

Identification of Special Impacts—Economic and Fiscal. Special economic and fiscal effects of nuclear repository siting arise from the unique characteristics of these facilities. The special economic effects of a repository will depend in large measure on the perceptions of area residents of the hazards associated with such a project. Possible effects include diminished activity in other basic sectors such as agriculture and recreation and the forestalling of other types of resource or industrial development. Investment in the local trade and service sector could be discouraged if the repository is perceived as a negative influence on long-term growth prospects for the area. Special fiscal effects could be experienced if the repository leads to unusual costs for state or local governments (e.g., for providing escorts for waste shipments or for preparing evacuation plans). Fiscal impact assessment for repositories also must include consideration of the tax-exempt status of these facilities.

Review of Impact Assessment Techniques

A detailed review of economic and fiscal impact assessment techniques was conducted as part of the project activities for 1980 (Murdock and Leistritz, 1980). This analysis indicated that the export base (economic base) theory is the basis for all economic impact assessments and that the principal methodological alternatives include various forms of export base employment or income multiplier models and input-output models. This analysis also reveals that, in selecting an economic assessment technique for repository siting, considerable attention should be given to the needs of decision makers in impacted areas. Greater emphasis should be placed on assessing the distribution of impacts over time, among jurisdictions, and among groups in the local population. Models that take project expenditure patterns into account and provide disaggregated impact projections should be used, and methods to reduce requirements for primary data collection must be developed.

Fiscal impact assessment techniques generally involve attempts to relate governmental revenues to changes in economic activity (e.g., income, sales, property value) and to relate costs to changes in the population served. Important considerations in developing a fiscal assessment technique for repository evaluation include:

1. Capability to address temporal and jurisdictional distribution of costs and revenues
2. Capability to evaluate effects of alternative approaches to service provision and financing
3. Transferability of the technique among potential sites in several states

Future Work Planned

Long term goals of the project include:

1. Producing a set of validated, reliable, and field-tested methods and processes for ensuring effective assessment and mitigation of the effects of repository siting
2. Responding to the time dimensions required for assessment processes and procedures
3. Maintaining the flexibility to respond to the issues and questions that arise within the socioeconomic dimensions of the siting process

The general sequence of the project from the 1981 fiscal year to the 1984 fiscal year will be as follows:

- FY 1981—model, methodology, and process design and development
- FY 1982—initial model, methodology, and process field assessments
- FY 1983—final model, methodology, and process assessments
- FY 1984—development of final assessment and mitigation manuals, programs, and other deliverables

During the 1981 fiscal year, major efforts related to economic and fiscal impacts will include:

1. Completion of an interactive computerized model for projecting economic, demographic, public service, and fiscal impacts of the construction and operation of nuclear repositories
2. Design of socioeconomic impact mitigation strategies for use in repository siting

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Part II Repository Development and Design

Rock Mechanics in the National Waste Terminal Storage Program

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The overall objective of the rock mechanics program of the Office of Nuclear Waste Isolation is to predict the response of a rock mass hosting a waste repository during its construction, operation, and postoperational phases. The operational phase is expected to be 50 to 100 yr; the postoperational phase will last until the repository no longer poses any potential hazard to the biosphere, a period that may last several thousand years. The rock mechanics program is concerned with near-field effects on mine stability, as well as far-field effects relative to the overall integrity of the geologic waste isolation system.

To accomplish these objectives, the rock mechanics program has established interactive studies in numerical simulation, laboratory testing, and field testing. The laboratory and field investigations provide input to the numerical simulations and give an opportunity for verification and validation of the predictive capabilities of the computer codes. Ultimately the computer codes will be used to predict the response of the geologic system to the development of a repository. 3 ref. 5 fig.

Introduction

Rock mechanics technology is required to design, license, construct, operate, and decommission mined repositories for disposal of high-level radioactive waste. Repository sites are subjected to two principal perturbations: (1) shaft sinking to repository level and excavation of the repository itself, and (2) thermal loading introduced by the radioactive decay of the waste. The first of these can be handled for the most part by careful design and use of applicable experience from civil and mining engineering. Because of the effects of the thermal loading, however, existing experience by itself is not sufficient to assure long-term isolation of the waste.

This paper is a summary of the rock mechanics research being conducted as part of the National Waste Terminal Storage (NWTS) program. We have purposely kept the technical detail in this paper to a minimum. Those who wish to pursue the details of the NWTS rock mechanics research should refer to the rock mechanics plan (Wigley et al., 1980). The research includes theoretical, laboratory, and field investigations to determine the rock properties and develop and calibrate the mathematical models required to predict the rock behavior for design studies, long-range predictions, and performance assessments of the repositories.

The immediate objective of this rock mechanics research is to provide the technology required to produce a credible conceptual design for repositories in several rock types by 1985. To reach this objective, we must progress in an orderly and planned manner from analytical or theoretical considerations in model development to laboratory and bench-scale testing and in situ testing. Two concerns must be resolved along the way:

1. How does the thermal-mechanical-hydrological response of the host rock to repository-induced perturbations affect the stability of the rock mass on a canister-, room-, and repository-scale?
2. How does the thermal-mechanical-hydrological response of the rock mass to repository-induced perturbations affect the long-term performance of the waste isolation system?

The first question relates to the design of the underground facility, i.e., the near-field, near-term performance. The answer to this question determines acceptable design parameters—including depth, areal thermal loading, extraction ratio, disposal room geometry and layout, support requirements, ventilation requirements. etc.—necessary to maintain the operational safety of the facility. The second question refers to the long-term performance of the waste isolation system. The answer to this question will provide the predicted hydrological data, over time, for the scenario and consequent performance assessment analyses.

Discussion

In developing the rock mechanics research program, several detailed technical questions that identify the gaps in our present knowledge of rock mechanics have been identified and, for simplicity, are grouped into five "umbrella" questions:

1. What are the important properties of the rock, and how does it behave as a function of time, temperature, and stress? This question relates to the quantification of physical properties of the rock mass and the determination of the in situ stresses and time-dependent constitutive relationships for the rock. The information obtained will be incorporated into computer codes, verified against test data, and used in the design and performance assessment of geologic disposal systems.

2. How do the repository-induced changes affect the behavior of the discontinuous rock mass, and how can the fractured rock mass best be characterized? This question addresses the fact that rock is generally a fractured medium and its behavior is dominated by the fracture system. Thus, to understand and model the behavior of a repository in fractured rock, we must first understand and model the behavior of the fractured rock system. This includes determining and characterizing the extent of the existing fracturing in the rock mass and describing the effect of repository-induced stress changes on the mechanical behavior of and groundwater flow through the fractured rock mass.

3. How can rock mass behavior be modeled? Both continuum and fractured rock models are being developed to predict the behavior of the rock mass. These models must consider not only questions 1 and 2 but also that there is a size effect in the behavior of rock and that adequate verification of the models must be obtained to provide confidence. Of course, for relative simplicity and for economy, models drawing on established continuum mechanics solutions would be preferred. However, analytical, laboratory, bench-scale, and field test comparisons may possibly indicate that continuum solutions do not adequately represent rock mass behavior. Development of fractured rock models is being pursued on a parallel track to counter this eventuality.

4. How do we confirm that a repository site responds as expected to the repository-induced perturbations? The answer to this question involves consideration of other issues, such as:

- What types of and how many field tests are required to establish confirmation of the site behavior?
- Does satisfactory instrumentation exist for the required field testing?
- To what extent can test results be transferred from site to site and rock to rock?

These issues will be reexamined frequently during the course of the research program, but the current thinking can be summarized as follows:

- Two independent sets of field tests are required—one to obtain the basic phenomenological understanding and one to verify the models.
- Some existing instruments are satisfactory; others will be obtained by ongoing research and development.
- Basic phenomenological understanding and testing methodology can be applied from one rock type or site to another, but in situ confirmation tests will probably be required for each rock type and site.

5. *What is the impact of the waste package on the rock mechanics program?* The waste isolation system consists of three functionally distinct but interacting systems—waste package, repository, and site. It is believed that modifications in the waste package have relatively little effect on the rock mechanics program. Technical requirements for resolving rock mechanics issues remain essentially unchanged because the functional requirements for the repository and site subsystems remain basically unchanged. The most significant effect is the accommodation of the package features into the repository design.

In late 1979 the Earth Science Technical Plan Rock Mechanics Subgroup of the Department of Energy and the U. S. Geological Survey evaluated the status of various technical elements required for the development of a repository in each host rock. We updated that evaluation to early 1982 for this paper. The revised evaluation is shown on Figs. 1 through 4 and is summarized in the following subsections.

Basalt. Additional research and development is required on discontinuity stress-deformation relationships, discontinuity stress-flow relationships, size effects, and thermal-mechanical-hydrological modeling, especially for discrete-fracture groundwater flow models. As model development progresses, verification will be achieved by laboratory, bench-scale, and field tests and by benchmark problems. The model must be completely verified before additional full-scale tests are initiated. The Near-Surface Test Facility (NSTF) will provide basic phenomenological understanding of the behavior of basalt in relation to repository-induced perturbations and will provide methodology that can be used to develop a suite of in situ confirmation tests. No new tests will be designed for the NSTF until the maximum benefit has been derived from the existing testing program.

Granite. Additional research and development are required on discontinuity stress-deformation relationships, discontinuity stress-flow relationships, size effects, and thermal-mechanical-hydrological modeling, especially for discrete-fracture groundwater flow models. Field testing in granite, as part of the basic phenomenological understanding and model development, has been completed (Stripa, Sweden) or is under way (Climax and the experimental mine at the Colorado School of Mines). Evaluation and study of these field test results will continue. Further field tests are not justified

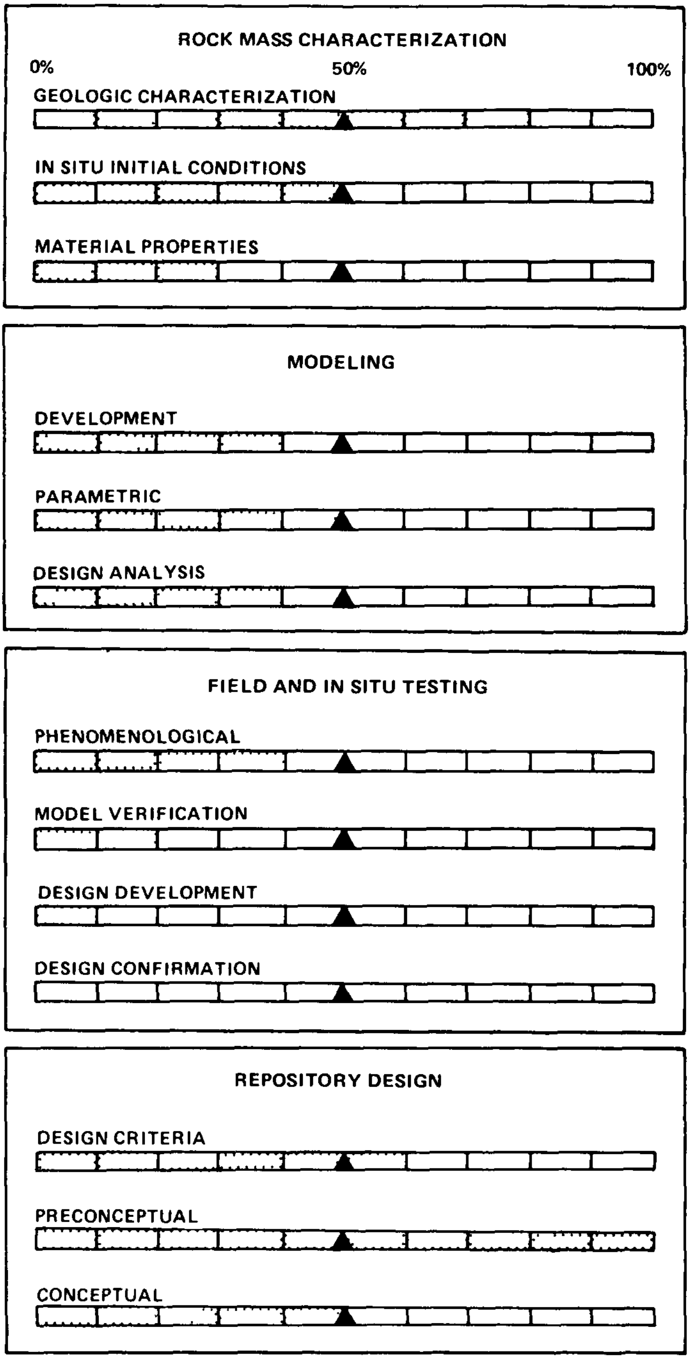


Figure 1 Status of technical elements required for development of a repository in basalt.

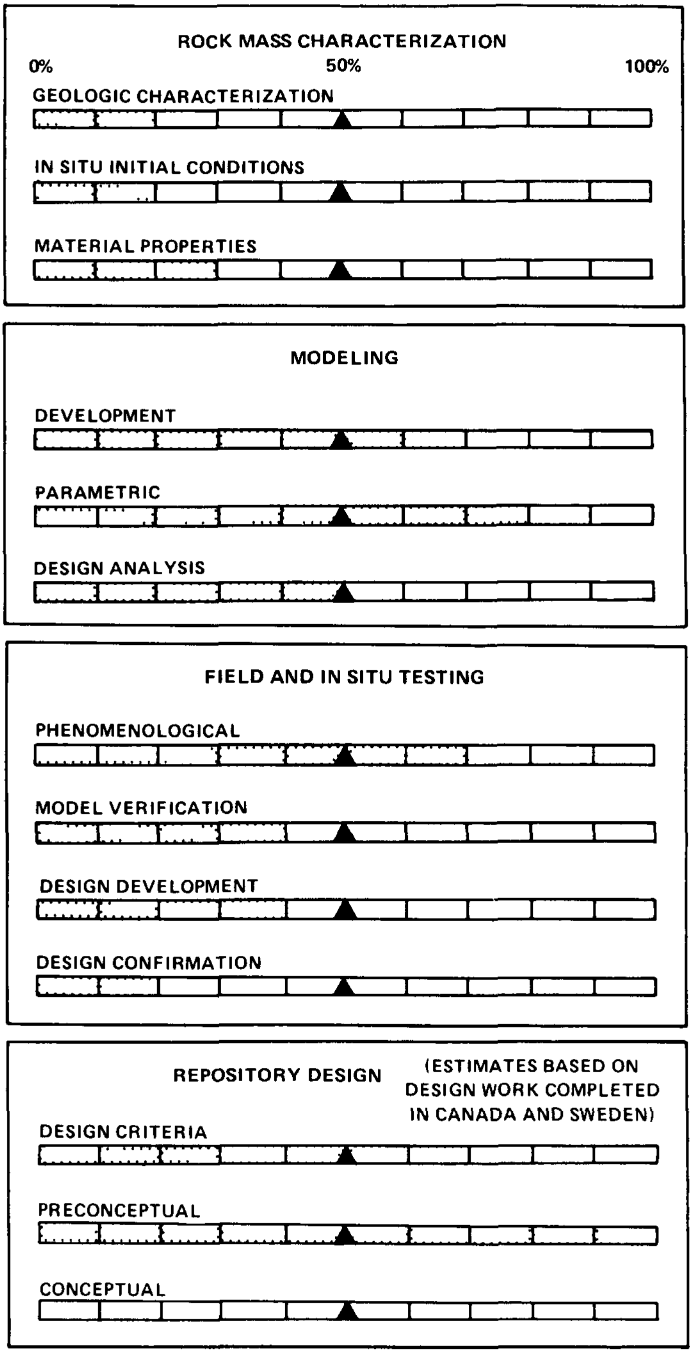


Figure 2 Status of technical elements required for development of a repository in granite.

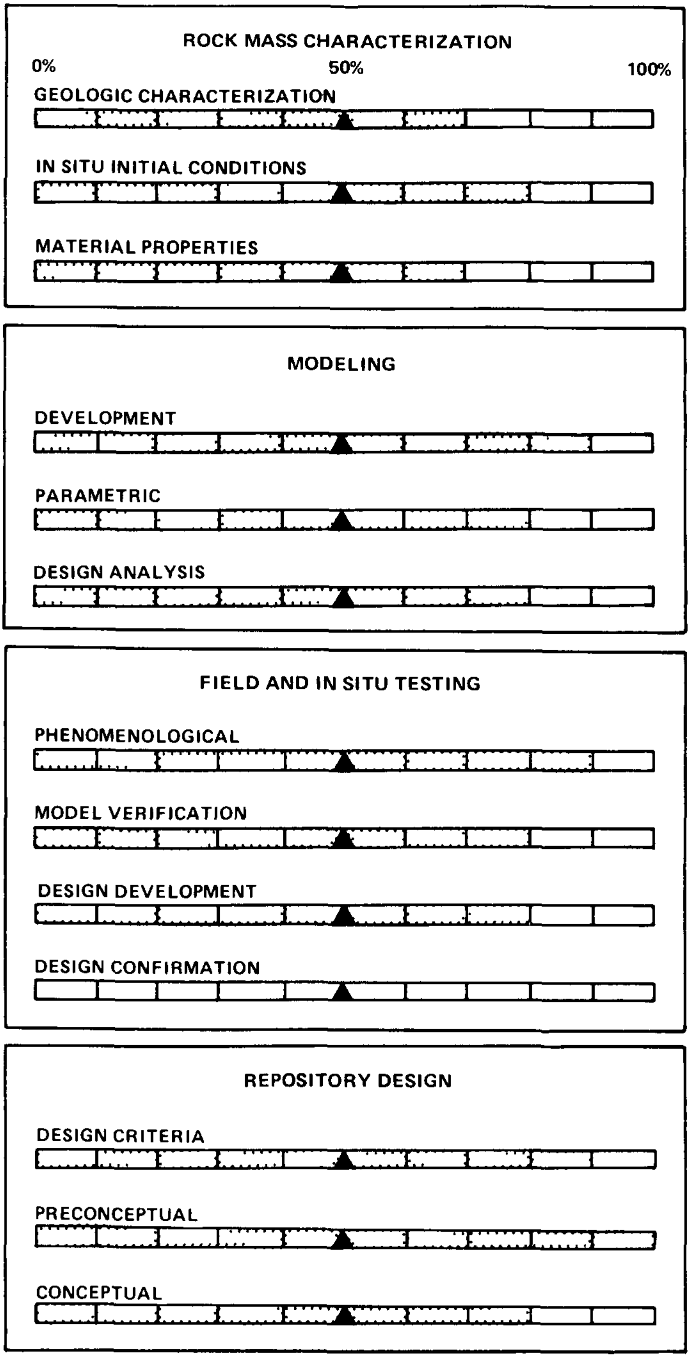


Figure 3 Status of technical elements required for development of a repository in salt.

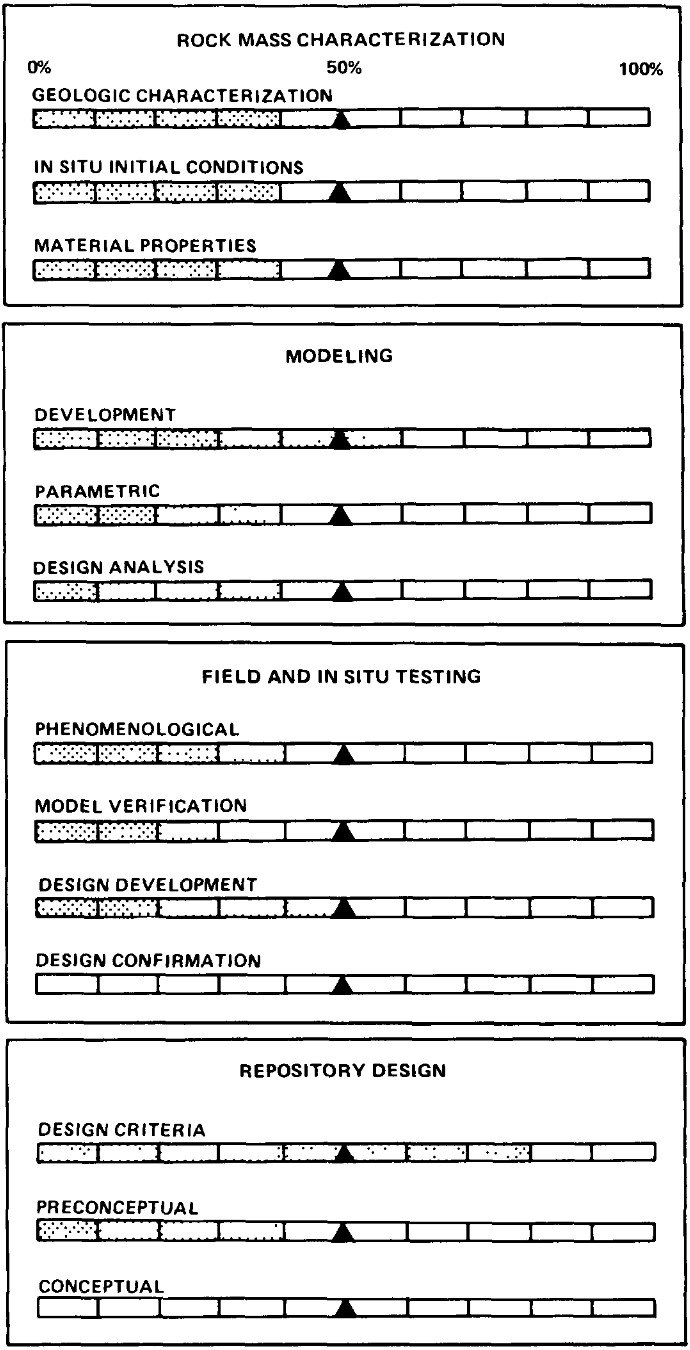


Figure 4 Status of technical elements required for development of a repository in tuff.

until maximum benefit has been gleaned from these tests. Since both basalt and granite are discontinuous crystalline rocks, the rock mechanics technology developed for one can be applied, perhaps with some modification, to the other.

Salt. At the present time basic phenomenological understanding, model development and verification, and field testing are further advanced for salt than for any other candidate rock type. Some additional effort will be made to improve creep predictions, especially over long time periods. The emphasis of the work on salt will be toward completion and documentation of work in progress, however, so that the manpower and resources can be applied to the other candidate rock types. Thus, unlike that for the other media, the technology for salt has progressed past the site identification and selection process. Salt has been shown to be a viable medium; the other candidate host rocks have not.

Tuff. Basic phenomenological understanding, model development, and verification for tuff are behind basalt and granite and far behind salt. Thus to bring tuff rock mechanics technology to the level required by 1985 will require a concentrated effort. Much of the technology developed for basalt and granite can be applied to tuff, but tuff can behave like a crystalline hard rock or a plastic rock, depending on the degree of welding. Consequently, strong emphasis will be placed on laboratory and bench-scale testing to determine the constitutive behavior of tuff. Field testing at G-tunnel, an existing tunnel in tuff on the Nevada Test Site, will also be actively pursued.

On the basis of the work by the Rock Mechanics Subgroup, the work done in preparing the NWTS rock mechanics plan (Wigley et al., 1980), and the present level of understanding of rock mechanics, rock types can be ranked from most to least understood:

1. Salt
2. Granite
3. Basalt
4. Tuff

The rock mechanics plan includes a detailed discussion of the critical milestones [39 have been identified (see Fig. 5)] for resolving the outstanding technical questions for each rock type.

In Fig. 5, the basic phenomenological understanding encompasses the development of both generic and site-specific data bases. These data will be used to formulate constitutive relationships describing the material's response to various loadings. Both the physical properties data base and the constitutive relationships will be incorporated into the numerical models. The constitutive relationships are essential in the coupling of the thermal-mechanical-hydrological phenomena.

The models will be verified against laboratory, bench-scale, and field data, as well as through code-to-code comparisons. Since the behavior of the host rock in the field is the ultimate concern of the NWTS rock mechanics program, the field test data bases are extremely important in both developing and verifying the numerical models. The data base from Project Salt Vault is an important test for all thermo-, visoelastic codes. This data base continues to increase our understanding of the response of salt to thermomechanical loadings. As the entire data base from the Avery Island heater experiments is disseminated and analyzed throughout the NWTS technical community, our understanding of salt's behavior will be substantially increased. The Stripa data base is extensive also, and, as it is analyzed, significant progress will be made on basic phenomenological understanding, model development, and model verification for granite.

The field tests at the Near-Surface Test Facility will contribute to the basic phenomenological understanding of basalt. The Climax Spent Fuel Test and the Colorado School of Mines tests will provide additional generic hard-rock data bases from which the NWTS rock mechanics program will benefit. The field tests at G-tunnel will provide the first test data base for tuff. Confirmation of the capability of numerical codes to reproduce the response of a rock mass in the field to thermomechanical loadings is essential to the development of a technically conservative waste isolation system. Consequently, field tests are an important element in the NWTS rock mechanics program.

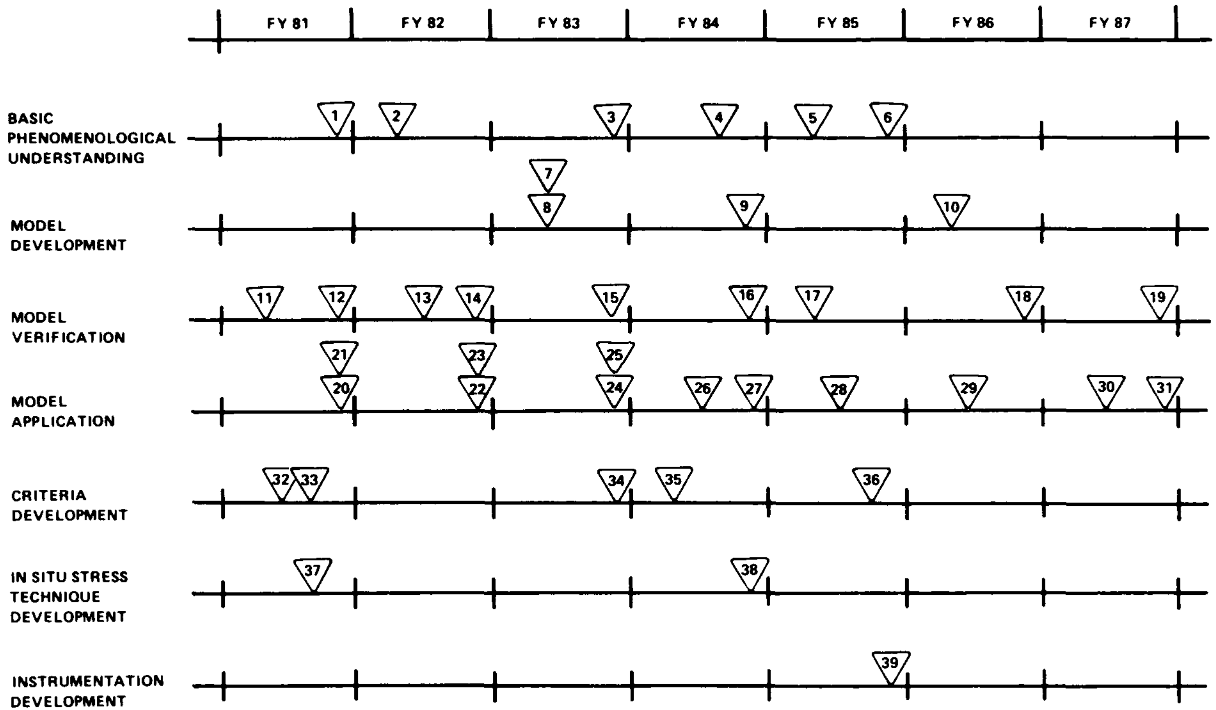
The models developed in this program, applied in predesign analyses, will aid the designer by establishing the problems that are likely to be encountered in developing a repository in different rock types. They will also be applied in various design phases before and during the construction of the facility. First, the initial conditions at the site are defined. After the rock mass has been characterized and the waste package has been defined, the canister-scale analysis is performed to ensure that none of the design bases have been violated. The design will be iterated until the optimum configuration for canister spacing has been established. The room-scale analysis is then performed. Again, the design bases cannot be violated for any point in time over the functional lifetime of the repository. Consequently, the design will be iterated until the optimum room and pillar layout has been established. Finally, the repository-scale analysis will be performed. The design will again be iterated until all the design bases have been met over their respective functional lifetimes. The models will then be applied in the performance assessment of the entire waste isolation system. The performance assessment application will be incorporated into the safety assessment reports. This approach has been thoroughly described in a series of papers on technical conservatism (Carbiener, Hall, and Matthews, 1980; Monsees, Wigley, and Carbiener, 1981).

Criteria will be developed and will be available to the designer so that the analysis described can be performed. The various phases of criteria development are described in Fig. 5.

In situ stress measurement techniques will be developed and necessary field instrumentation will be improved before the first waste isolation system is developed.

Summary

The rock mechanics technology required to successfully design, license, construct, operate, and decommission a geologic waste isolation system was described. The NWTS rock mechanics program will concentrate on basic phenomenological understanding, model development, and verification in the near term. Conceptual design studies will be undertaken concurrently with these investigations. Further field and in situ tests are required for ultimate model verification and site confirmation. These will be conducted in the early construction phase of repository development. The research and development effort will also be directed toward understanding and interpreting results from field tests, such as those at Stripa and Project Salt Vault and from the Near-Surface Test Facility, Avery Island, Climax, G-tunnel, and the Colorado School of Mines. A more complete understanding and interpretation of these tests must be obtained. These will be incorporated into our basic phenomenological understanding and modeling techniques and, thereby, serve to support conceptual design. Performance assessment investiga-



FY 81
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BASIC PHENOMENOLOGICAL UNDERSTANDING
MODEL DEVELOPMENT
MODEL VERIFICATION
MODEL APPLICATION
CRITERIA DEVELOPMENT
IN SITU STRESS TECHNIQUE DEVELOPMENT
INSTRUMENTATION DEVELOPMENT
1 2 3 4 5 6 7 8 9 10 11 12 13 14 15 16 17 18 19 20 21 22 23 24 25 26 27 28 29 30 31 32 33 34 35 36 37 38 39

BASIC PHENOMENOLOGICAL UNDERSTANDING

1. GENERIC TIME-DEPENDENT CONSTITUTIVE RELATIONSHIP ESTABLISHED FOR BEDDED AND DOMAL SALT (9/81)
2. GENERIC SIZE-EFFECT STUDY COMPLETED (1/82)
3. GENERIC PHYSICAL PROPERTIES DATA BASE ESTABLISHED (9/23)
4. GENERIC DISCONTINUITY STRESS-FLOW CONSTITUTIVE RELATIONSHIP ESTABLISHED (6/84)
5. GENERIC DISCONTINUITY STRESS-DEFORMATION CONSTITUTIVE RELATIONSHIPS ESTABLISHED (1/85)
6. INITIAL SITE-SPECIFIC PHYSICAL PROPERTIES DATA BASE ESTABLISHED (9/85)

MODEL DEVELOPMENT

7. INITIAL THERMAL–MECHANICAL–HYDROLOGICAL DISCRETE FRACTURE MODEL AVAILABLE (3/83)
8. INITIAL FULLY COUPLED THERMAL–MECHANICAL–HYDROLOGICAL CONTINUUM MODEL AVAILABLE (3/83)
9. DECISION ON APPLICABILITY OF CONTINUUM MODELS TO REPRESENT DISCONTINUOUS MEDIA (9/84)
10. FULLY COUPLED THERMAL–MECHANICAL–HYDROLOGICAL MODELS OPERATIONAL (1/86)
11. INITIAL ANALYSIS OF AVERY ISLAND HEATER TESTS COMPLETE (3/81)

MODEL VERIFICATION

12. INITIAL ANALYSIS OF STRIPA TESTS COMPLETE (9/81)
13. FINAL ANALYSIS OF PROJECT SALT VAULT COMPLETE (4/82)
14. FINAL ANALYSIS OF CLIMAX MINE BY EXPERIMENT COMPLETE (9/82)
15. FINAL ANALYSIS OF AVERY ISLAND HEATER TESTS COMPLETE (9/83)
16. FINAL ANALYSIS OF STRIPA TESTS COMPLETE (9/84)
17. INITIAL ANALYSIS OF THE NEAR-SURFACE TEST FACILITY TESTS COMPLETE (1/85)
18. INITIAL ANALYSIS OF CLIMAX SPENT FUEL TEST COMPLETE (9/86)

MODEL APPLICATION

19. FULLY COUPLED THERMAL–MECHANICAL–HYDROLOGICAL MODELS GENERICALLY VERIFIED (9/87)
20. FINAL ROOM-SCALE STABILITY REPORTS ISSUED FOR BEDDED AND DOMAL SALT AND BASALT (9/81)
21. GENERIC CONCEPTUAL DESIGN ISSUED FOR BASALT (9/81)
22. FINAL ROOM-SCALE STABILITY REPORT ISSUED FOR GRANITE (9/82)
23. FINAL REPOSITORY-SCALE STABILITY REPORTS ISSUED FOR BEDDED AND DOMAL SALT AND BASALT (9/82)
24. FINAL REPOSITORY-SCALE STABILITY REPORT ISSUED FOR GRANITE (9/83)
25. FINAL ROOM-SCALE STABILITY REPORT ISSUED FOR TUFF (9/83)
26. INITIAL DESIGN ASSESSMENTS OF CANDIDATE LOCATIONS COMPLETE (4/84)
27. FINAL REPOSITORY-SCALE STABILITY REPORT ISSUED FOR TUFF (9/84)
28. GENERIC CONCEPTUAL DESIGN ISSUED FOR GRANITE (3/85)
29. GENERIC CONCEPTUAL DESIGN ISSUED FOR TUFF (2/86)
30. SITE-SPECIFIC PRELIMINARY DESIGN COMPLETE (3/87)
31. SITE-SPECIFIC PRELIMINARY SAFETY ASSESSMENT REPORT ISSUED (9/87)

CRITERIA DEVELOPMENT

32. DRAFT GENERIC REPOSITORY CRITERIA ISSUED (2/81)
33. FAR-FIELD ROCK FAILURE CRITERIA COMPLETE (6/81)
34. NEAR-FIELD ROCK FAILURES CRITERIA COMPLETE (9/83)
35. MEDIUM SPECIFIC BASE-LINE CRITERIA AND SPECIFICATIONS COMPLETE (1/84)
36. SITE-SPECIFIC REPOSITORY DESIGN CRITERIA COMPLETE (6/85)

IN SITU STRESS TECHNOLOGY DEVELOPMENT

37. COMPARISON OF HYDRAULIC FRACTURING TECHNIQUES AND LEEMAN CORE TECHNIQUE COMPLETE (6/81)
38. IN SITU STRESS TECHNIQUES ESTABLISHED (9/84)

INSTRUMENTATION DEVELOPMENT

39. RELIABLE STRESS GAUGES DEVELOPED (9/85)

Figure 5 Major milestones in the NWTS rock mechanics program.

tions will be conducted for each site recommended for repository development.

Finally, research efforts on salt will be brought to a documented level of completion. This is necessary to provide the resources and manpower for work on the other rock types, especially tuff, so that they can be brought to a comparable level of understanding by early 1986.

Acknowledgment

This paper was presented, in abbreviated form, at the 1980 National Waste Terminal Storage Program Information Meeting, December 1980, and appeared in the proceedings of that meeting. The paper was then expanded, essentially as presented here, for the American Nuclear Society Topical Meeting, Waste Management '81, Tucson, AZ, February 1981. For this volume, we updated the paper to early 1982.

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Influence of Specimen Size on the Creep of Rock Salt

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Triaxial compression creep data for Avery Island dome salt are analyzed to determine the influence of specimen size on creep deformation. Laboratory experiments were performed on 50- and 100-mm-diameter specimens in the temperature range from 25 to 200°C and the axial stress difference range from 2.5 to 31.0 MPa. The strain-vs.-time data from each test are divided into transient and steady-state components. Results of statistical analysis of these data show that transient creep of the small specimens is a stronger function of stress, temperature, and time than is transient creep of the larger specimens. Analysis of the steady-state data show no size effect, however.

Introduction

Constitutive laws that describe the rate-dependent deformation of salt are formulated by first performing laboratory experiments to generate a data base and then fitting a model to these data so that strain can be predicted as a function of thermomechanical history (stress, temperature, and time). These laboratory-based constitutive laws are used to predict the deformation of large-scale structures, including nuclear waste repositories. Whether or not the constitutive laws obtained from small, laboratory-scale specimens can adequately represent the material behavior at large, field-scale structures must be assessed. The influence of specimen size on strength has been investigated for hard rocks (Heuze, 1980), as well as for salt (Stickney, 1977). The strength of hard rocks decreases as specimen size increases; thus strength values determined in the laboratory provide unsafe estimates of strength at the field scale. Stickney found, however, that the strength of salt increases with specimen size, and thus laboratory strength values are conservative estimates of strength at the field scale. The purpose of this study is to determine whether there is a size effect in the creep of salt and, if there is such an effect, whether laboratory-based creep laws are conservative.

A series of triaxial-compression creep experiments was performed on specimens of natural rock salt to determine the influence of specimen size on the creep of salt. Both small (50-mm-diameter) and large (100-mm-diameter) cylindrical specimens were tested. The data from these experiments are fitted to simple constitutive laws to determine values of the parameters in the laws. Fits are made for each specimen size. The parameter values and confidence intervals obtained for each specimen size are compared to determine whether the parameter values are equal at the specified level of confidence of the confidence intervals.

Analysis of the results showed that, at the 95% confidence level, transient creep has a significant size effect, but steady-state creep has no size effect. Transient creep is a stronger function of stress, temperature, and time in the small specimens than in the larger specimens. Therefore transient creep laws derived from laboratory data will overpredict deformations at the field scale because, for a given stress and temperature, the strains in the small specimens are greater than those in the large specimens.

Test Specimens and Jacketing

The natural rock salt tested in this program was obtained from the Avery Island Mine, which is operated by International Salt Company at Avery Island, LA. The three mining tiers, 150-, 210-, and 275-m (500-, 700-, and 900-ft) levels, are situated near the center of a salt diapir. A geological cross section of the Avery Island dome is shown in Fig. 1. The 150-m level was mined in the early 1900s; the 210-m level was depleted within the last decade; and, at present, mining is in progress at the newly opened 275-m level.

RE/SPEC initiated field activities in the Avery Island Mine at the 210-m level in 1976. As a result of these field studies, core was available from three 0.4-m- (16-in.-) diameter holes.

The specimens used in this work were recored from these field samples using a vertical milling machine, a thin-walled diamond-impregnated bit, and saturated brine solution. They were cut to a length equal to approximately twice their diameter on a band saw. The ends were polished flat and parallel with a diamond wheel in the vertical mill. Finished specimens had tolerances of ± 0.08 mm on the diameter, and the ends were finished parallel to within ± 0.005 mm.

Avery Island dome salt is composed primarily of strain-free halite grains, ranging in size from 2.5 to 15 mm with an average diameter of 7.5 mm. The rock salt is unusually clean. It has little anhydrite and clayey material along grain boundaries and few negative and anhydrite crystals within the halite grains.

Two of the three field cores were examined for fabric using the technique described by Carter (1977), and the results are presented in Fig. 2. Four parallel thin sections were required to measure the 300 a-axes in 100 halite crystals from the first core (Fig. 2a), and six sections were needed to determine the orientations of 225 a-axes in 75 grains of halite from the second core (Fig. 2b). The fabric patterns of the two cores are virtually identical, with concentrations on moderate a-axes parallel to the core axis (vertical) spreading into girdles. Fabrics of this type and strength are typical of many dome salts (Carter, 1977; Schwerdtner, 1968). Concentrations of a-axes parallel to the core (stress) axis and in a plane normal to it ensure that most crystals are oriented favorably for dodecahedral [110] glide, the primary slip system in halite.

Triaxial tests require that the specimen be jacketed to prevent contamination by the confining fluid. Initially, the specimen is inspected for any pits that may have resulted from coring. These pits are filled with a silicone sealant to prevent jacket intrusion under pressure. The specimen is then placed between two cylindrical steel platens. The salt-to-steel interfaces are lubricated with a molybdenum disulfide dry lubricant. A length of Viton tubing is then slipped over the assembly and sealed to each platen by inserting a Viton O ring between the jacket and the platen and placing a silicone sealant above and below the O ring. Steel tie wires, wrapped around the jacket above and below the O-ring location, compress the jacket against the platens. This assembly is allowed to dry overnight before it is placed in the load frame for testing.

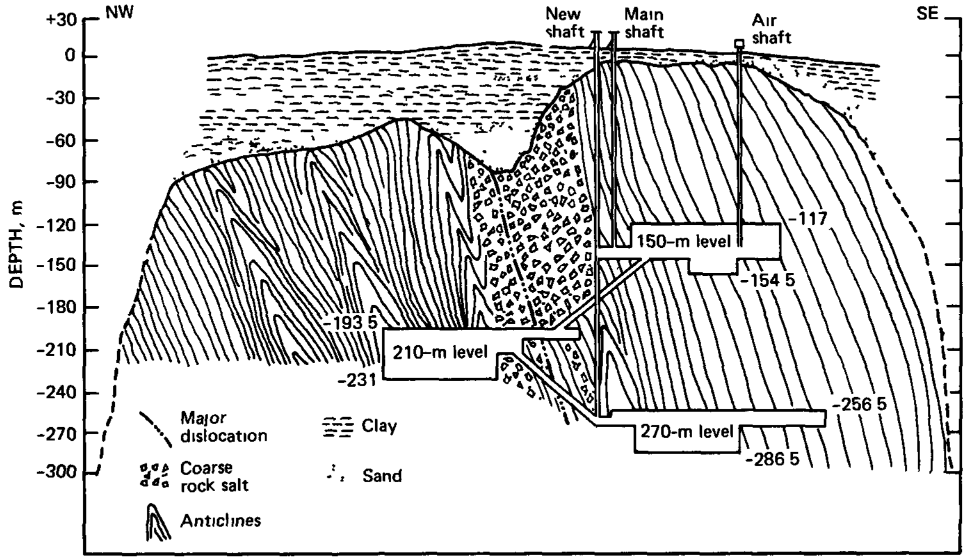

Figure 1 Avery Island dome geological cross section. (Data from Fairchild and Jenks, 1978.)

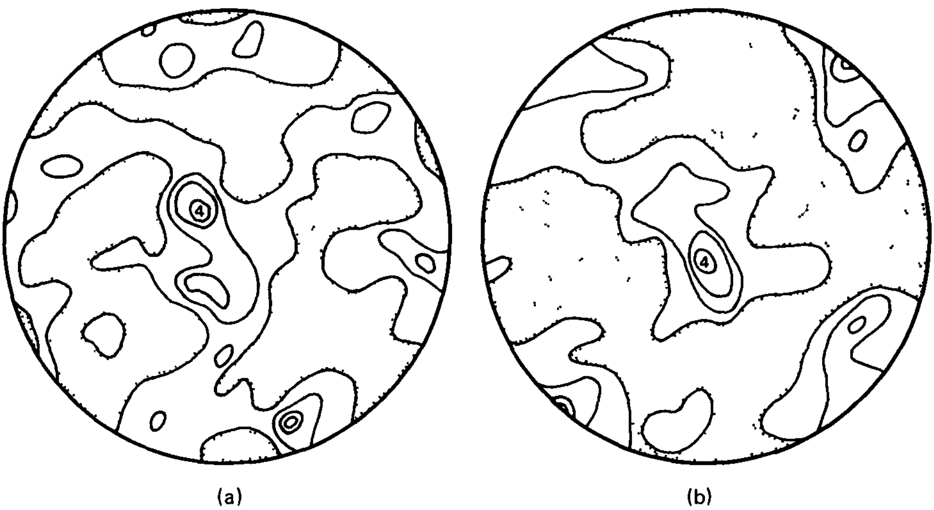

Figure 2 Fabrics of cores from Avery Island dome salt. Equal area, lower hemisphere projections with contours at intervals of 1% per 1% area.

Testing Machines and Procedure

The machines used to perform the creep experiments have been described previously in great detail (Hansen and Mellegard, 1980; Mellegard, Senseny, and Hansen, 1981). Six testing machines were used in this study—two small machines for testing the 50-mm-diameter specimens and four larger machines for testing the 100-mm-diameter specimens. All six machines operate identically and have similar capabilities.

Figure 3 shows a schematic of one of the larger testing machines. The machines have conventional triaxial cells, and triaxial experiments are conducted in compression: $\sigma_1 > \sigma_2 = \sigma_3$, where σ_1, σ_2, and σ_3 are the maximum, intermediate, and minimum compressive principal stresses. The apparatus is equipped with transducers that measure axial and lateral stress, axial and lateral displacement, and temperature. Axial and lateral stress are monitored electronically with a load cell and an in-line pressure transducer, respectively. These measurements can be readily compared with mechanical pressure gauge readings. Axial displacement is measured by a linear variable differential transformer (LVDT). Lateral displacement is measured and the lateral pressure is controlled with a dilatometer (a reversible motor system). The volume change of the confining liquid in the triaxial cell is a measure of the average lateral strain (Crouch, 1970; Wawersik, 1975). Temperature is measured by a thermocouple located in the wall of the pressure vessel, which also provides feedback to the temperature controller.

The load cells and gauges are accurate to ± 0.07 MPa. The confining pressure transducers are accurate to ± 0.04 MPa. Resolution of the axial and lateral displacement measurements is approximately equivalent to strains of ± 30 and ± 50 $\mu\epsilon$, respectively. Temperatures are controlled to within ± 1°C from room temperature to 200°C.

At the beginning of an experiment, a jacketed specimen is placed in the triaxial cell, and the cell is filled with silicone oil (the confining fluid), heated to the test temperature, and allowed to stabilize. Next the specimen is subjected to a hydrostatic stress equal to the confining pressure. The axial stress difference is then applied at a rate of 1.15×10^{-2} MPa/s.

Time zero for the creep test is the time at which the desired axial stress difference is reached. A computer collects data at time intervals determined by the rate of deformation of the specimen. If the specimen is deforming rapidly, data can be collected as often as once per minute; whereas, if the specimen is deforming slowly, data are collected only once per hour. At intermediate rates of deformation, data are collected at some interval whose length is proportional to the rate of deformation.

The Cauchy (or true) stress is maintained constant during the test by increasing the load on the specimen as it deforms and its cross sectional area increases. The deformations measured are used to calculate natural (or logarithmic) strain. The stress and strain measures used in this study are Cauchy stress and natural strain.

Test Matrixes

The influence on creep of specimen size is studied for both transient and steady-state deformation. Since many tests were short-duration experiments, not all those used in the study of transient creep were used in the study of steady-state creep. Tables 1 and 2 give the transient and steady-state test matrixes, respectively. The data analyzed are all from stage I experi-

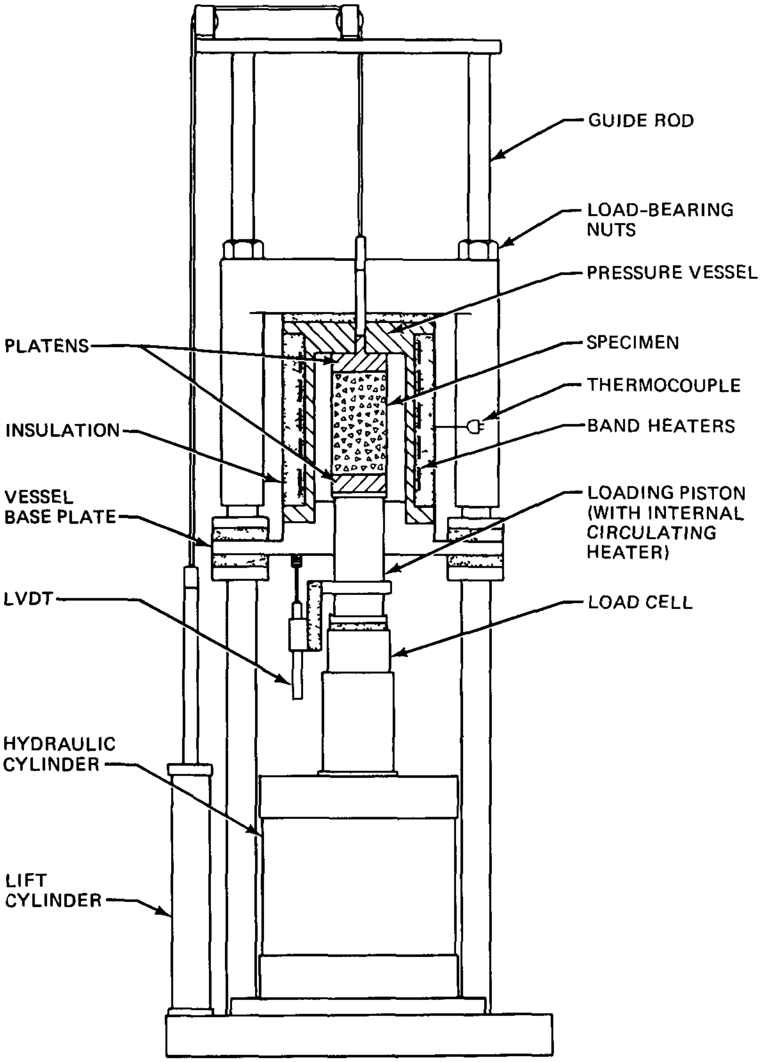

Figure 3 Test machine load frame.

ments, i.e., experiments in which the specimen had no previous loading in the laboratory.

Each "X" in the transient creep matrix (Table 1) corresponds to a pair of tests, one performed on a large specimen and one on a small specimen. There are twenty-one pairs of tests in the matrix. The axial stress differences range from 2.5 to 31.0 MPa; the temperatures range from 297 to 473°K; and confining pressures range from 0.7 to 20.7 MPa.

Since not all tests included in Table 1 reached steady state, only eleven X's appear in Table 2, the steady-state test matrix (again, each X corresponds to a matched pair of tests). The ranges of stress and temperature are nearly the same as for the transient creep matrix.

Table 1 Test Matrix for Study of Transient Creep

Axial stress difference, MPa	Temperature, 297°K			Temperature, 343°K				Temperature, 373°K			Temperature, 473°K		
	Confining pressure, MPa			Confining pressure, MPa				Confining pressure, MPa			Confining pressure, MPa		
	3.5	13.8	20.7	0.7	6.9	13.8	20.7	3.5	13.8	20.7	3.5	13.8	20.7
2.5												X	
5.0									X				
5.5											X		X
6.9	X												
10.3	X	X	X	X	X	X	X	X	X	X	X		
18.1									X				
20.7	X	X							X				
31.0	X												

Table 2 Test Matrix for Study of Steady-State Creep

Axial stress difference, MPa	Temperature, 297°K			Temperature, 343°K				Temperature, 373°			Temperature, 473°K		
	Confining pressure, MPa			Confining pressure, MPa				Confining pressure, MPa			Confining pressure, MPa		
	3.5	13.8	20.7	0.7	3.5	13.8	20.7	3.5	13.8	20.7	3.5	13.8	20.7
2.5												X	
5.0									X				
5.5											X		X
6.9													
10.3	X			X	X		X		X		X		
15.0									X				
20.7													
31.0													

Data Analysis

Test data comprise axial creep strain measured at discrete times while the stress and temperature were held constant (±0.07 MPa, ±1.0°C). Individual test results are given elsewhere (Hansen and Mellegard, 1980; Mellegard, Senseny, and Hansen, 1981). The data from each test were divided into transient and steady-state parts. In the transient regime the specimen deforms with a monotonically decreasing strain rate; whereas in the steady-state regime specimen deformation occurs at a constant strain rate. The transition between these two regimes is very smooth and, consequently, difficult to locate.

The usual procedure for finding the steady-state regime is to lay a straightedge along the most linear part of the plot of strain vs. time. All the data that can be reasonably approximated by the straight line is taken to be in the steady-state regime. The steady-state strain rate is simply the slope of the straight line, and the transition between transient and steady-state regimes is determined to be the point at which the data deviate unacceptably from the straight line. This procedure is subjective, and results can vary when different persons perform the analysis.

An algorithm has been developed that eliminates some of this subjectivity. The procedure is to search the log ϵ, log t data for an interval in which a linear least-squares fit gives $(\Delta\log \epsilon/\Delta\log t) = 1 \pm \eta$ and a correlation coefficient of $r^2 \geqslant r_0^2$. This interval is taken to be the steady-state regime. The parameter η determines how closely the strain-vs.-time curve in the interval must approximate a straight line. These two parameters, which are chosen subjectively by the analyst, give a quantitative measure of the subjectivity introduced. A purely objective algorithm would set $\eta = 0$ and $r_0^2 = 1$. The steady-strain rate is found from a linear least-squares fit to the strain-vs.-time data in the steady-state interval. For this study, the values $\eta = 0.01$ and $r_0^2 = 0.985$ were used.

Since data were not collected at equal time intervals in all tests, the transient creep data from each experiment were fitted to the equation

$$\epsilon = kt^n \tag{1}$$

where ϵ is axial creep strain, t is time, and k and n are fitting parameters. A nonlinear regression algorithm* was used to make these fits, and the fits obtained were excellent. These equations were then used to compute strains at times desired for the analysis. The durations of the transient regimes of the two tests in any given pair were usually not equal, because one of the specimens did not reach steady-state. Therefore, the longer test was truncated so that both tests would have equal transient duration. Test duration does vary from pair to pair, however.

All statistics calculations for this report were performed using the computer routines provided by International Mathematical Statistical Libraries, Inc. (1978). They are available on most computer systems.

Transient strain data from the 21 tests on the small (50-mm-diameter) specimens and the 21 similar tests on the large (100-mm-diameter) tests were fit to a constitutive law having the form

$$\epsilon = A\ \sigma^m\ t^n\ T^p \tag{2}$$

where

ϵ	=	axial creep strain
σ	=	axial stress difference
t	=	time measured from the start of creep
T	=	absolute temperature
A, m, n, and p	=	parameters whose values are determined by fitting the data

The constitutive law (Eq. 2) can be transformed to a linear equation by taking the logarithm of both sides:

$$\ln \epsilon = \ln A + m\ln \sigma + n\ln t + p\ln T \tag{3}$$

In this form, linear regression techniques can be used to determine the constants ln A, m, n, and p when the errors in ln ϵ are minimized. The strain data used to fit the constitutive law are taken at equal intervals of ln t. Each test contributes 100 points.

Table 3 gives the values of ln A, m, n, and p and their 95% confidence intervals for both the 50- and 100-mm-diameter specimens. The parameter values are not equal for the

Table 3 Transient Constitutive Parameter Estimates and Their 95% Confidence Intervals

Parameter	ln A*			m			n			p		
Specimen size	Lower confidence limit	Best estimate	Upper confidence limit	Lower confidence limit	Best estimate	Upper confidence limit	Lower confidence limit	Best estimate	Upper confidence limit	Lower confidence limit	Best estimate	Upper confidence limit
50 mm diameter	-57.89	-56.586	-55.279	3.03	3.09	3.15	0.427	0.434	0.442	6.51	6.72	6.92
100 mm diameter	-47.983	-46.553	-45.127	2.69	2.76	2.82	0.396	0.404	0.413	5.02	5.25	5.47

**Natural logarithm, stress difference in MPa, time in seconds, and temperature in °K.*

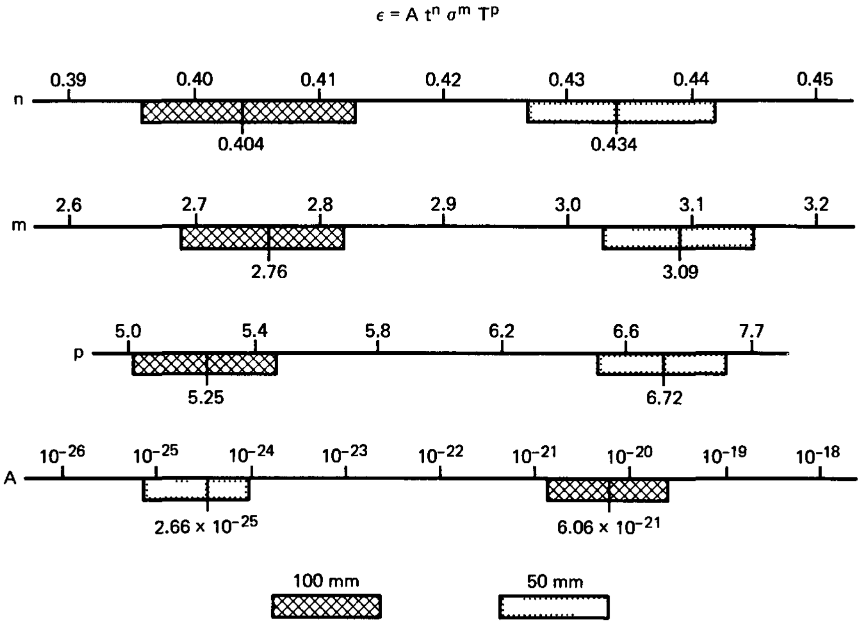

Figure 4 Transient constitutive parameters and their 95% confidence intervals for the large and small specimens.

two specimen sizes. Furthermore, the values for one specimen size do not fall within the 95% confidence intervals determined for the other. In fact, as shown in Fig. 4, the confidence intervals do not even overlap. This is sufficient evidence that the constitutive parameters are statistically different for the two specimen sizes.

The constitutive parameter values obtained for the two specimen sizes show that creep of the small specimens is a stronger function of stress, time, and temperature than is creep of the large specimens. Figures 5 and 6 show creep strains computed from the two constitutive laws. At low temperature (Fig. 5) and low stress (Fig. 6), the strain in the small specimens is less than that in the large specimens; at high temperature and high stress, it is greater. Both figures show that the strain in the small specimens increases more rapidly with time than does that in the large specimens.

The constitutive laws obtained by substituting the parameter values from Table 3 into Eq. 2 differ from those presented previously for Avery Island salt (Hansen and Mellegard, 1980; Mellegard, Senseny, and Hansen, 1981) for two reasons. First, only stage I data are fit in this study; whereas stage II and stage III* data were included in previous studies. Secondly, the previously derived constitutive parameters were not obtained by multiple linear regressions. In the first of those studies (Hansen and Mellegard, 1980), the stress difference and temperature exponents were determined from linear least-squares fits to the data at one specific time (t = 100 hr). In the second study (Mellegard, Senseny, and Hansen, 1981), the constitutive param-

**In multistage creep tests a single specimen is subjected to a sequence of constant stress and temperature conditions. The first condition is referred to as stage I, the second as stage II, etc.*

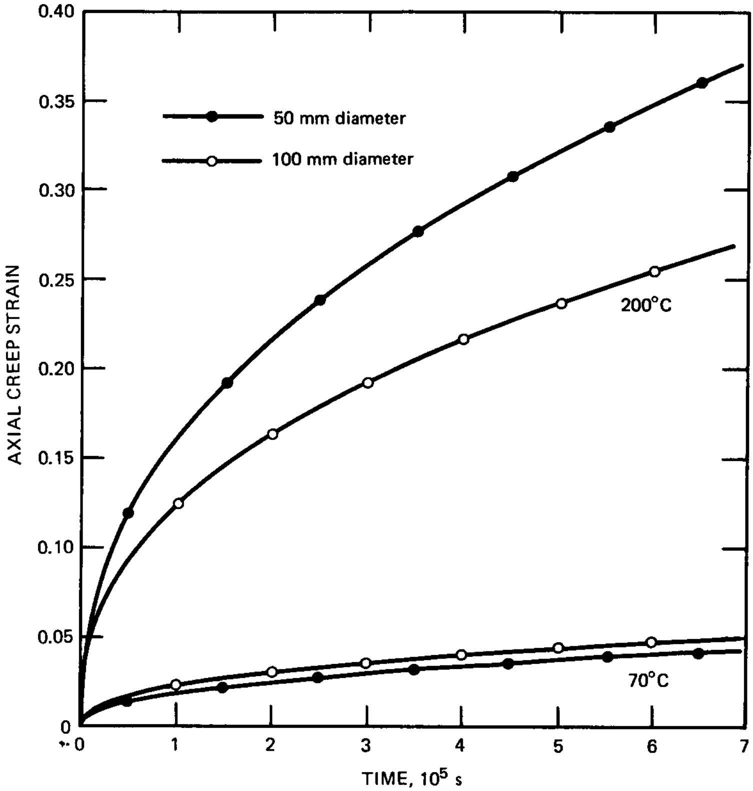

Figure 5 Transient creep strain as a function of time for 50- and 100-mm-diameter specimens of Avery Island salt at σ = 15 MPa.

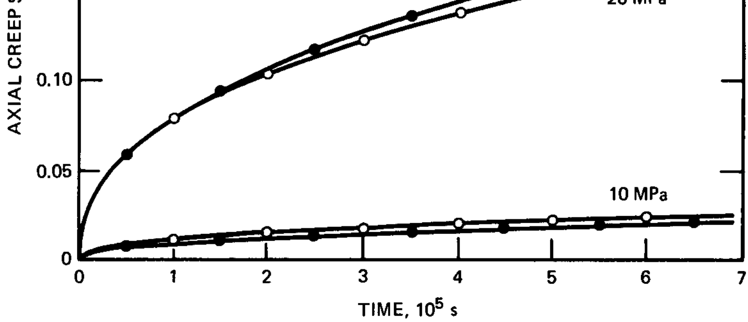

Figure 6 Transient creep strain as a function of time for 50- and 100-mm-diameter specimens of Avery Island salt at T = 100°C.

eters were determined using nonlinear least-squares regression.

The steady-state strain rates from the tests on both small and large specimens were fitted to the equation

$$\dot{\epsilon}_{ss} = B\sigma^{r} \exp(-Q/RT) \quad (4)$$

where
$\dot{\epsilon}_{ss}$ = axial steady-state strain rate
σ = axial stress difference
T = absolute temperature
B, r, and −Q/R = parameters whose values are determined by fitting the data

The parameter −Q/R is usually interpreted as the negative of the ratio of activation energy for rate-controlling deformation mechanisms to the universal gas constant.

Equation 4 can be transformed to linear form by taking logarithms of both sides:

$$\ln \dot{\epsilon}_{ss} = \ln B + r\ln \sigma - \frac{Q}{RT} \quad (5)$$

Linear regression techniques provide values of the parameters ln B, r, and -Q/R and their 95% confidence limits, shown in Table 4. The parameter values are nearly equal, and the parameter values for one specimen size are contained within the 95% confidence intervals that surround the parameter values for the other specimen size. Figure 7 shows graphically that, at the 95% confidence level, steady-state creep is not influenced by specimen size.

Discussion

The presence of a statistically significant influence of specimen size in transient creep but not in steady-state creep is perplexing. During transient creep, while the microstructure is changing, specimen size is important, but during steady-state creep, when the microstructure is constant, specimen size does not influence the creep rate. This implies that evolution of the microstructure (hardening) depends on specimen size. One possible explanation is presented in terms of difference in the volume-to-surface-area ratios of the two specimen sizes. As the specimen hardens during transient creep, the dislocation density increases, but, during steady-state creep, the dislocation density remains constant. Since the ratio of the surface area to the volume is twice as large for the 50-mm-diameter specimens as for the 100-mm-diameter specimens, a larger fraction of the dislocation lines intersects the free surface of the small specimens. Although a specific model cannot be presented, the specimen size effect during transient creep may be related to the percentage of new dislocations that intersect the free surface. During steady-state creep, when the dislocation density is constant, the percentage of dislocations that intersect the free surface is also constant, and no size effect is observed. The influence of specimen size on the transient creep of salt appears to be explainable in terms of some undefined surface phenomenon. Other observations (Hirth and Lothe, 1968) indicate that mechanical properties of salt and other ionic crystals may depend on surface phenomena. For example, dissolution of the surface of a specimen decreases its brittleness (the Joffe effect), and the adsorption of surface active molecules decreases the specimen's strength and hardness (the Rehbinder effect).

Another data set, collected from the same size specimens of bedded salt from southeastern New Mexico, was analyzed to determine the influence of specimen size on creep (Herrmann,

Table 4 Steady-State Constitutive Parameter Estimates and Their 95% Confidence Intervals

Parameter	ln B*			r			−Q/R		
Specimen size	Lower confidence limit	Best estimate	Upper confidence limit	Lower confidence limit	Best estimate	Upper confidence limit	Lower confidence limit	Best estimate	Upper confidence limit
50 mm diameter	-13.309	-10.569	-7.832	2.84	3.80	4.76	4650	5900	7140
100 mm diameter	-14.197	-11.220	-8.247	2.72	3.87	5.03	4400	5790	7170

Natural logarithm, stress difference in MPa.

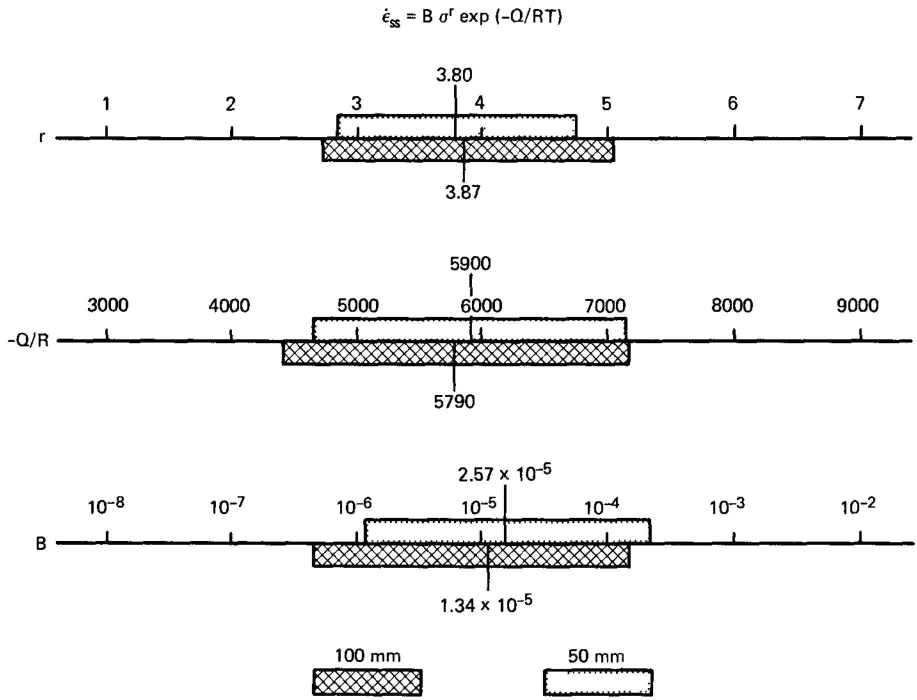

Figure 7 Steady-state constitutive parameters and their 95% confidence intervals for the large and small specimens.

Wawersik, and Lauson, 1980a, 1980b). These investigators concluded that there is no difference between creep of the small (50-mm-diameter) specimens and that of the large (100-mm-diameter) specimens. The test procedures and testing machines used for the study of southeastern New Mexico bedded salt are very similar to those used in this study of dome salt from Avery Island. In fact, the 50-mm-diameter specimens of both salts were tested on the same machines, and the larger specimens of both salts were tested on machines of identical design. The difference in the conclusions of these two studies could result from differences either in behavior of the two salts or in the way the data were analyzed. The difference in behavior of the two salts cannot be assessed. However, constitutive laws fitted to data from 50-mm-diameter specimens of both salts are very similar (Hansen and Carter, 1980). The difference in analysis techniques probably accounts for the opposing conclusions.

I determined parameter values for the creep laws and their 95% confidence limits for each specimen size. If the difference between the parameter values was small in comparison with the width of the 95% confidence limits, no size effect was postulated. If the difference between parameter values was large, however, a specimen size effect was apparent. In the study of the southeastern New Mexico salt, dummy variables were used in the constitutive model. When the model was fitted to the data, these variables were not found to be significant. This led to the conclusion that specimen size does not influence creep of salt. Results obtained from studies that use dummy variables must be interpreted carefully. If specimen size is not included in the model properly, results of fitting the model to the data may indicate that specimen size is not important even though it does influence creep. A very simple example illustrates this point. Suppose the data are distributed according to $y = +(x^2 - r^2)^{1/2}$, the equation of a semicircle. If the model $y = mx + b$ is fit to the data, the variable x will not be significant even though the value of y is uniquely determined by the value of x. Therefore, even though specimen size was found to be not significant for southeastern New Mexico, the result could be an artifact of the procedure used to analyze the data.

Summary and Conclusions

A study is presented of the influence of specimen size on creep of salt. Laboratory experiments were performed in triaxial compression on 50- and 100-mm-diameter specimens of dome salt from Avery Island, LA. The data were separated into two parts, one for the transient regime and the other for the steady-state regime, by using a special algorithm that objectively determines the steady-state strain rate and the transition between the two regimes. Transient creep strains for both specimen sizes were fitted to a power-law creep model. When parameter values and their determined 95% confidence limits were compared, specimen size was found to have a significant influence. Transient creep of the smaller specimens is a stronger function of stress, temperature, and time. Transient creep laws based on laboratory data overpredict the short-term deformation expected at the stresses and temperatures in the repository near field.

A similar procedure was followed in comparing the steady-state strain rates obtained from tests on the two sizes of specimens. Specimen size does not influence steady-state creep. The presence of a size effect for transient creep but not for steady-state creep possibly results from a surface phenomenon that influences creep only during the transient regime when the specimen is hardening because the dislocation density is increasing.

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Seismic Effects on Underground Openings

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Seismic risk to subsurface facilities cannot be judged by applying intensity ratings derived from surface effects of an earthquake. Mines have operated for a substantial period of time in some of the most seismically active regions in the world, but quantitative data on subsurface damage caused by earthquakes are few. This fact itself attests to the lessened effect of earthquakes in the subsurface. More damage is reported in shallow tunnels near the surface than in deep mines. In mines and tunnels, large displacements occur primarily along preexisting faults and fractures or at the entrance to these facilities. Data indicate that vertical subsurface structures, such as wells and shafts, are less susceptible to damage than are surface facilities.

For some types of subsurface facilities, such as those for long-term storage, the damage caused by displacement is more significant than that caused by shaking.

Computer calculations for several types of faults indicate that displacements decrease rapidly with distance from a fault but can either increase or decrease as a function of depth, depending on the type and geometry of the fault. For long, shallow, strike-slip faults, displacement decreases rapidly with depth. For square strike-slip faults and for dip-slip faults, displacement does not decrease as markedly with depth.

Numerical modeling techniques were used to determine the conditions required for seismic waves generated by an earthquake to cause instability to an underground opening or create fracturing and joint movement that would lead to an increase in the permeability of the rock mass. Three different rock types (salt, granite, and shale) were considered as host media for the repository located at a depth of 600 m. Special material models were developed to account for the nonlinear material behavior of each rock type. The sensitivity analysis included variations in the in situ stress ratio, joint geometry, and pore pressures, and the presence or absence of large fractures. Three different sets of earthquake motions were used to excite the rock mass.

The methodology applied was found to be suitable for studying the effects of earthquakes on underground openings. In general, the study showed that moderate earthquakes (up to 0.41 *g*) did not cause instability of the tunnel or major fracturing of the rock mass; however, a tremor with accelerations up to 0.95 *g* was amplified

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around the tunnel, and fracturing occurred as a result of the seismic loading in salt and granite. In situ stress is a critical parameter in determining the subsurface effects of earthquakes but is nonexistent in evaluating the cause for surface damage. In shale with the properties assumed, even the moderate seismic load resulted in tunnel instability.

These studies are all generic in nature and do not abrogate the need for site and design studies for specific facilities.

Introduction

The potential seismic risk for an underground nuclear waste repository will be considered in evaluating its ultimate location. Methods of assessing the seismic response of surface facilities are well developed. Particularly for the licensing of nuclear power reactors, extremely detailed studies of seismicity and seismic effects have been made. For the surface facilities of a nuclear waste storage repository, the methodology of seismic assessment will follow that already developed for other purposes. In the past, however, there has been little motivation to study subsurface seismic effects of earthquakes. The mining industry generally pays little attention to seismic effects because other hazards are much greater and earthquakes are not common on a daily basis.

Numerous published statements indicate that subsurface damage caused by earthquakes is not nearly as severe as surface damage in the same location caused by the same earthquake. Some statements suggest there is no damage to facilities at depths greater than 152 m (500 ft). Stories are common of miners coming out of the mine and expressing disbelief when told of an earthquake that did damage or was felt at the surface.

The purposes of this study were to quantify these qualitative statements to the extent possible and to provide an understanding of the behavior of underground openings in response to seismic stress. Three studies are reported here. The purpose of the first was to present a compendium of data collected from existing literature. The purpose of the second was to analyze the displacement field around an underground opening caused by seismic stress, on the basis that displacement is of greater significance to a nuclear repository than vibratory motion for long-term considerations. The purpose of the third study was to use numerical modeling techniques to analyze the potential damage to an underground opening caused by a variety of conditions pertinent to a nuclear repository, including seismic stress.

Earthquake Damage to Underground Facilities

Scattered through the available literature are statements to the effect that, below a few hundred meters, shaking and damage in mines are less than that at the surface. Data for decreased damage underground have not been completely reported and explained, however.

To assess the seismic risk for an underground repository, Pratt, Hustrulid, and Stephenson (1978) established a data base and evaluated the potential for seismic disturbance. To develop this data base, the investigators searched the literature to document the damage or nondamage to underground facilities caused by earthquakes and to evaluate the significance of the data. A number of reports listed earthquake damage to underground structures, such as mines and tunnels, but these were primarily of a qualitative nature. Displacements associated with four major earthquakes in several parts of the world were documented by Duke and

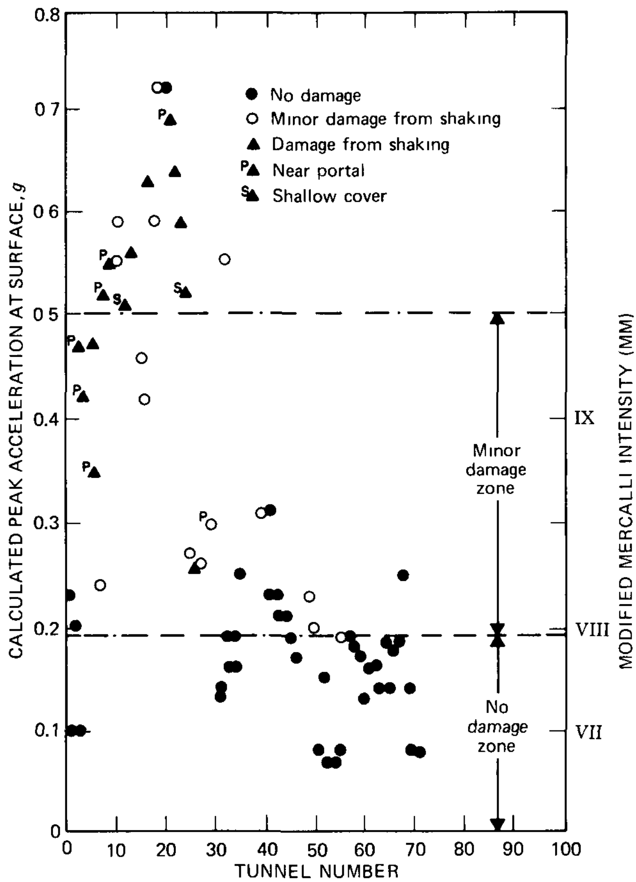

Figure 1 Calculated peak acceleration at the surface and associated tunnel damage. (Data from Rozen, 1976; Dowding, 1978.)

Leeds (1961). More recently, the effect of earthquakes on shallow tunnels, primarily in the United States, has been collected and analyzed (Rozen, 1976; Dowding, 1978). In addition to these data, a large number of individual reports have indicated both damage and nondamage resulting from earthquakes of magnitudes greater than 5 (Jagger, 1923; Kawasumi, 1964; National Research Council, 1964; Bolt et al., 1975; Steinbrugge and Moran, 1954; Stevens, 1977).

Other sources of potential information were investigated, e.g.:

- More complete and recent data from foreign sources in earthquake prone areas, such as Japan
- Data from mining operations where earthquakes are initiated by the mining process
- Results from the nuclear events at the Nevada and Alaskan Test Sites, as well as Plowshare experiments. These tests provide the most quantitative data in the near-field environment. The tests were well instrumented and may assist in evaluating and establishing damage criteria with respect to the seismic spectrum resulting from an earthquake.

Tunnels and Shallow Underground Openings. Data on the seismic stability and behavior of

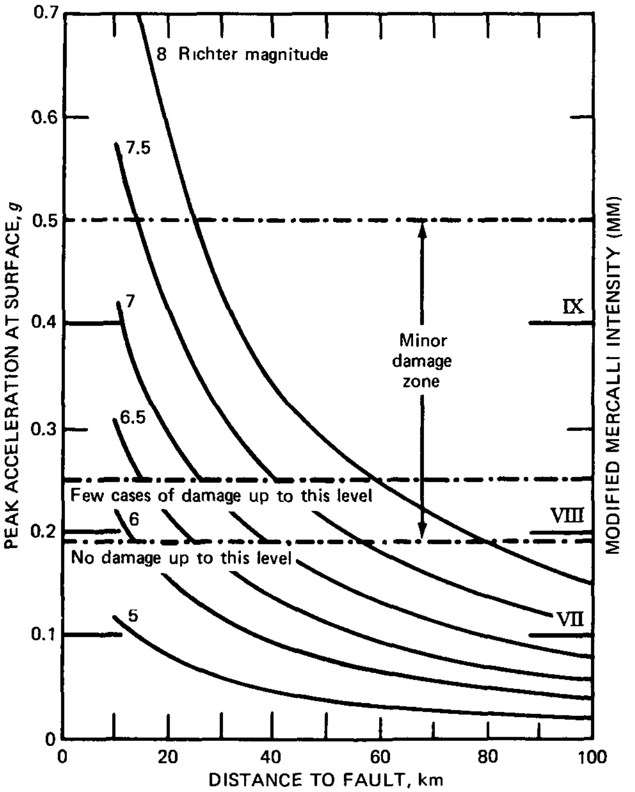

Figure 2 Accelerations, Modified Mercalli intensity, and associated tunnel damage. (Data from Rozen, 1976; Dowding, 1978.)

shallow underground openings are very well summarized by Rozen (1976) and Dowding (1978). Observations from 71 tunnels responding to earthquake motions were compared. Dynamic behavior was compared with intensity and magnitude as a function of distance. The studies compared calculated accelerations at the ground surface with tunnel damage and showed that tunnels are less susceptible to damage than are surface structures or facilities. Peak acceleration at the surface of less than 0.2 *g* did not damage the tunnels; between 0.2 and 0.5 *g*, damage was only minor; and damage was significant only above 0.5 *g* (Fig. 1). Most of the damage that occurred was located near a portal. Richter magnitude and Modified Mercalli (MM) intensity were correlated with acceleration for various cases (Fig. 2). Large accelerations are correlated with large magnitudes and high intensities. At any one specific site, calculations of surface accelerations were based on the earthquake magnitude and epicentral distance through attenuation laws developed by McGuire (1974). No reduction was made for attenuation with depth.

Dowding (1978) summarized that (1) tunnels are more stable than structures located on the surface, and (2) critical frequencies are lower for large underground chambers than for tunnels because of the increase in the size of underground chambers.

The conceptual designs of many underground facilities indicate that rooms will probably be ~10 m in diameter. The critical frequency calculated from Rozen's data for underground openings of this size is 150 Hz, and therefore threshold damage would not occur unless the facility was relatively near the earthquake epicenter. Perhaps most important, the primary cause of failure of underground excavations is relative movement along preexisting faults or at the portal of the tunnel, which is located at ground surface.

Duke and Leeds (1961) reviewed information on tunnel damage, as well as some mine damage caused by earthquakes, and drew the following conclusions:

1. Severe tunnel damage appears to be inevitable when the tunnel is crossed by a fault fissure that slips during the earthquake.
2. In tunnels away from fault breaks, shaking may cause severe damage to linings and portals where construction is of marginal quality and the structure is located in the epicentral region of strong earthquakes.
3. Well-constructed tunnels outside the epicentral region but away from fault breaks can be expected to suffer little or no damage in strong earthquakes.
4. Within the usual range of destructive earthquake periods, intensity of shaking below ground is less severe than that on the surface.

Mines and Other Deep Openings. Reports on earthquake damage to underground mines have generally been qualitative in nature. Quantitative data, which are much more difficult to obtain, come primarily from a few sources. Most of the quantitative data are in the form of displacements or accelerations noted in mines in Japan, South Africa, and the United States.

Several Japanese investigators measured earthquake motion simultaneously at depth and at the surface. Nasu (1931) determined the ratio of displacements caused by earthquakes at the surface and in tunnels at depths up to 160 m. One of the most striking displacements was the 2.3-m transverse horizontal offset 0.6 m beyond a tunnel heading during the 1930 Tanna earthquake. Ratios of surface to depth displacement were 4.2, 1.5, and 1.2 for periods of 0.3, 1.2, and 4 s, respectively. The geology consisted of lake deposits at the surface and volcanic andesite and agglomerates at a depth of 160 m. Nasu concluded that underground motion may be one-fourth that at the surface.

Kanai and Tanaka (1951) measured accelerations at depths up to 600 m in copper mines in Paleozoic rock at Hitachi, but, unfortunately, recorded data were from small earthquakes. The ratio of surface maximum displacement to that at a depth of 300 m was about 6 : 1.

Iwasaki, Wakabayashi, and Tatsuoka (1977) obtained acceleration records to depths of 150 m below the surface during a 5-yr period from borehole accelerometers installed at four locations around Tokyo Bay. Three of the sites were in sands and clays, and one was in a siltstone. During the period of operation, data were obtained from 16 earthquakes, ranging in magnitude from 4.8 to 7.2. Iwasaki and co-workers concluded from analysis of the accelerations recorded in the boreholes at different depths that the distribution of maximum acceleration varies considerably with the change of soil conditions near the ground surface. Ratios of surface acceleration to that at the deeper layer (110 to 150 m) are about 1.5 at rocky ground, 1.5 to 3 at sandy ground, and 2.5 to 3.5 at very clayey ground. Although the acceleration values are smaller at deeper layers, frequency characteristics of underground seismic motions are close to those of the surface motions.

Information on earthquake damage from South Africa was obtained during discussions with U. S. Geological Survey (USGS) personnel. On Dec. 16, 1976, a damaging earthquake of magnitude 5.0 to 5.5 was recorded at Welkom, South Africa. The surface damage was extreme, with large structures failing. Displacements ≤10 cm were noted in the mine at a depth of 2.0 km. The focal depth of the earthquake was ~6 km.

In both the Rand Gold and the Orange Free State districts, studies were conducted to assess the relationship of acceleration, displacement, and frequency of earthquakes to magnitude during mining operations. These mines are up to 4 km deep. McGarr (1977) noted that shear displacements on the order of 5 to 10 cm, caused by stress redistribution, were associated with rock bursts of magnitude 2 to 3. These data are very important and, along with the data at Welkom, may give some indication of upper bounds of displacements near earthquake sources in these very hard rocks.

The USGS study of the Alaskan earthquake of 1964 (Eckel, 1970) stated that no significant earthquake damage was reported to underground facilities, such as mines and tunnels, although some rocks were shaken loose in places. Included in this analysis were reports of no damage in the coal mines of the Matanuska Valley, the railroad tunnels near Whittier, the tunnel and penstocks at the Eklutna hydroelectric project, and the Chugach Electric Association tunnel between Cooper and Kenai lakes. There were also no reports of damage to the oil and gas wells in and along Cook Inlet. These reports of no damage from the Alaskan earthquake are significant. This earthquake was one of the largest (M = 8.5) to occur in this century, and surface damage was extreme.

During the 1960 Chilean earthquake, one of the strongest earthquakes on record, miners in coal mines heard strange noises but felt no effects of the quake. Later examination of the mines, which extend under the ocean, showed several old faults but no new movement (Stevens, 1977).

Similar results were reported by Cooke (1970) for the Peru earthquake of May 31, 1970. This earthquake, of Richter magnitude 7.7, did no damage to 16 railroad tunnels totaling 1740 m under little cover in MM VII to VIII intensity zones. Also, no damage was reported to the underground works of a hydroelectric plant and three coal and two lead-zinc mines in the MM VII intensity zone.

Nuclear Events as Earthquake Simulators. The use of nuclear events as equivalent earthquake sources has been discussed by Rodean (1970) and Pratt, Hustrulid, and Stephenson (1978). The data from nuclear events can be useful in assessing the potential damage to underground facilities from earthquakes. The resulting velocities, accelerations, and displacements from nuclear events have been monitored carefully because of their importance to defense-related issues. In many cases the data are obtained at conditions like those near the hypocenter of the earthquake and, thus, more severe than would be anticipated from any earthquake affecting an underground facility. It should be possible, however, to place certain bounds on the maximum accelerations, velocities, and displacements expected from comparable earthquakes. This would be helpful in establishing damage criteria for potential earthquake damage.

At the outset, it is important to compare nuclear events with earthquakes to determine the scaling relationships between the two. An important point to make is that "comparable magnitude" indicates only that P wave signals from both earthquakes and explosions are of equal

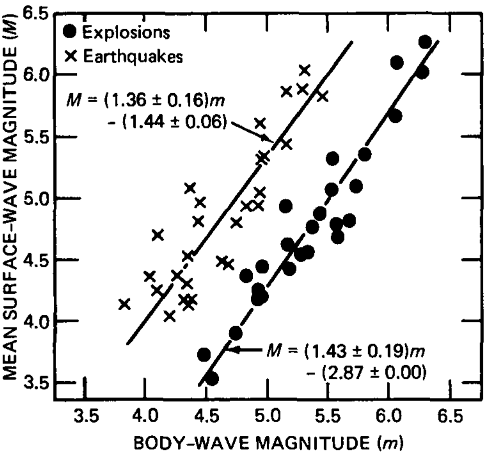

Figure 3 Mean surface-wave magnitude vs. body-wave magnitude for 28 earthquakes and 26 nuclear explosions in southwestern North America, as determined by Canadian measurements. (Data from Rodean, 1970.)

strength. Nuclear explosions tend to produce much weaker surface waves than do earthquakes of comparable body-wave magnitude (Fig. 3). As a consequence, the surface-wave energy associated with an earthquake of a given body-wave magnitude is on the order of 10 times that of an explosion of an equal body-wave magnitude (Rodean, 1970). Therefore an explosion of magnitude 5 does not have the same potential for causing ground motion damage at the surface as does an earthquake of magnitude 5. Pratt, Hustrulid, and Stephenson (1978) and Perret (1972) analyzed displacements, accelerations, and velocities from nuclear events in rock at depth and at the surface.

Wells. The damage to water and oil wells has been documented in a limited number of reports. Failure of water wells is caused primarily by sanding and silting, but in some instances there has been crushing, bending, or shearing of the casing as a result of differential movement of the surrounding rock. This latter mode of failure has also affected some oil wells. The damage to wells appears to be more of a near-surface phenomenon than one at depths of >100 m, except where the well crosses a fault.

Some damage to wells occurred during the earthquake on Feb. 9, 1971, in San Fernando, CA (Department of the Interior and Department of Commerce, 1971). Minor damage was reported to a few oil wells in the area, and all seven wells that supplied water to the city of San Fernando suffered damage during the earthquake, resulting in a severe water supply problem. Casings of oil wells in the greater Los Angeles area which cross faults have been ruptured by movement along the faults, but it is uncertain whether the movement is creep of a tectonic origin or settlement due to subsidence. Damage to wells in the San Joaquin Valley as a result of compaction of sediments, which is caused by the withdrawal of groundwater, is relatively common, but this damage is from aseismic causes.

The USGS documented the effects of the Alaskan earthquake (Mar. 27, 1964) on wells throughout most of Alaska and on the changes in water levels noted in the lower 48 states. Waller (1966) summarized the damage to wells in Alaska as mainly sanding or silting of the well or differential movement of the casing caused by movement of the surrounding rock. Three city wells and possibly one private well were damaged in Anchorage. Three city wells in Seward were damaged and rendered useless by ground movement and fissuring. In Valdez the casing of one well was sheared at a threaded joint 4.7 m below ground surface. No damage was reported to any of the oil and gas wells in and along Cook Inlet.

In general, the performance of wells during earthquakes is quite good, with the major damage resulting from bending, crushing, or shearing of the casing as a result of differential movement of the surrounding rock. The major damage appears to be to shallow wells in unconsolidated sediments and near the surface. There is very little damage to wells deeper than about 100 m, except where the well crosses a fault plane along which movement occurs.

Conclusions on Earthquake Damage to Underground Facilities. In summary, the damage to underground tunnels, mines, and wells does not have a large data base, especially with respect to measured displacement. The relation between velocity (and, thus, distance for M = 5, 6, and 6.5) and damage level has been summarized by Rozen (1976), however. Strong tensile and some radial cracking was noted at surface velocities of 152 cm/s, which would occur at distances of about 7 to 8 km during an earthquake of magnitude 6.5. Even at these levels, seismic damage would be negligible in competent rock.

The data for measured displacements as a function of depth are summarized in Table 1. Surface displacements range from at least 1 to 10 m, depending on geology, magnitude, etc., but decrease markedly with depth. Displacements of 25 cm or less have been measured 100 m deep in in situ rock masses, and displacements of <7 m were noted along preexisting faults. The data base below 500 m is almost negligible. The one data point from South Africa needs more detailed study of displacement, rock type, and local tectonic environment.

Acceleration and displacement data from nuclear explosions may give close-in upper-bound limits for large earthquakes.

Frequencies most likely to cause damage to subsurface facilities are significantly higher (50 to 100 Hz) than those (2 to 10 Hz) which cause damage to surface facilities.

Earthquake-Related Displacement Fields Near Underground Facilities

Damage to long-term underground facilities needs to be viewed in a somewhat different light than damage to surface facilities. Acceleration is generally considered to be the principal cause of surface damage, but a much higher level of acceleration is required to do subsurface damage. After the operational period of the waste repository, however, when it becomes passive, the type of damage done to surface structures is of little consequence. Of greater interest to the long-term facility is whether an earthquake could enhance the naturally low permeability of the rocks surrounding the repository. For this consideration, displacements caused by an earthquake were evaluated by Pratt, Zandt, and Bouchon (1979).

Block Motion Phenomena. Acceleration, velocity, and displacement as functions of distance and magnitude of earthquakes were discussed in detail by Pratt, Hustrulid, and Stephenson (1978). Observed fault displacements at depth from some of the larger earthquakes are summarized in Table 1.

Seismological Evidence for Block Motion. The seismological evidence for block motion comes from nuclear events, earthquakes, and large explosions. The evidence includes, for the large explosions, far-field ground motion recordings, the observation of surface faulting, and near-field ground motion recordings (Bache and Lambert, 1976; Archambeau and Sammis, 1970). Earthquakes occur when the local tectonic stress field increases beyond the failure strength of the rock mass. The stress release, which may cause block motion displacements, is probably caused by either a prestressed medium or the asymmetry of the source of the earthquake (Bache and Lambert, 1976).

Table 1 Displacements at Depth for Major Earthquakes

Location and year	Depth, m	Displacement, cm	Type	M
San Francisco (1906)	214	137	Shear existing fault	8.3
South Africa (1976)	2000	10		5.1
Japan (1930)	140	750	Horizontal	7.0
Japan (1930)	160	239	Horizontal	
Japan (1930)	160	51	Vertical	7.0
San Fernando (1971)	Surface	190	Vertical	6.5
Japan (1923)	76	25		8.16
Japan (1923)	50	<1		8.16
Kern County (1952)	50	<20		7.6
Kern County (1952)	73	<20		7.6

Simulation Experiments. Evidence of block motion exists from near-surface high-explosive tests (Blouin, 1979), including tests in sedimentary and igneous rock. The differential displacement, particle velocity, and potential displacements at actual and scaled ranges were measured for these high-explosive events. Block motion displacements observed in these events include joint block displacement, thrust block displacement, and fracture and bedding plane movements. Surface displacements up to several feet have been measured.

Models. Various analytical models were formulated to estimate potential displacements during loading as a result of explosions. Dai and Lipner (1977) developed models to assess various regions of interest for block motion displacements. These include the free-surface region, where thrust block model and surface block-dynamic response models are applicable, and areas at depth, where kinematic and incipient fault motion models are applicable. These models may be modified for an earthquake source, in terms of both geometry from the deep source and the nuclear-earthquake source energy ratios. The relative influence for a particular joint fracture or fault system will be important. Whether a shear failure occurs depends on the distance of the fault or discontinuity from the source, the orientation and continuity of the discontinuity, and the local in situ stress conditions. These considerations are obvious and must be addressed in making quantitative predictions concerning failure. In most practical instances, however, analytical solutions can be prohibitively complicated because of material inhomogeneity and nonlinear effects near the source in the presence of free surface.

An idealization of this problem, which has been solved analytically, is one of incipient fault motion caused by a spherical elastic wave in an infinite homogeneous isotropic medium (Johnson and Schmitz, 1976). Using reasonable frictional failure criteria, they calculated the failure surface of an arbitrary plane as a function of orientation and distance from an explosive source. Results from the analytical solution indicated that orientations near 60° (between the plane of the discontinuity and the plane containing the earthquake source and the point of interest) are most susceptible to failure and that the timing of the pulse arrival and the pulse shape affect joint failure. For very sharp pulses and angles less than 35°, failure will not occur no matter how close the joint or fault is to the source. These results should be transferable to earthquake sources.

Static Displacement Fields of Earthquakes. Since most of the data on static displacements associated with earthquakes is confined to surface observations, static deformation as a function of depth was calculated.

Surface Displacements. After the dislocation theory was developed, permanent surface deformation was used to infer the faulting parameters. A dislocation surface is a plane within an elastic medium across which there is a discontinuity in the displacement vector (i.e., a fault). Steketee (1958) used the theory of dislocations in a semi-infinite, isotropic, elastic medium as a mathematical model of faulting. Chinnery (1961) used some of Steketee's results to study the surface deformation around rectangular, vertical, strike-slip faults.

Accompanying the development of the theory was the accumulation of geodetic data on observed surface deformation associated with large earthquakes. One of the earliest earthquakes with well-documented surface displacements is the San Francisco earthquake of 1906. Horizontal displacements greater than 4 m were documented near Tomales Bay. Vertical displacements associated with the 1964 Alaskan earthquake were documented over an area of about 200,000 km^2, with maximum uplift averaging 3 m over a broad area.

For obvious reasons, most of the data on static deformation associated with earthquakes is confined to surface observations. Thus in the analytical studies the equations of the deformation field were often simplified by eliminating the depth-dependent term. Yet, that is exactly what is required for this study. Therefore a computer program was developed to provide the static deformation field around a fault as a function of depth.

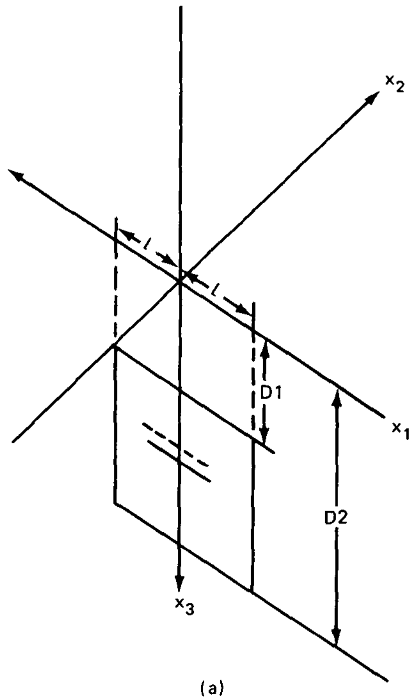

Figure 4 (a) Coordinate system for calculating displacement fields:

$$U_k = \frac{1}{8\pi\mu} \iint_\Sigma \Delta u_i \, \omega_{ij}k \, V_j d \sum$$

$$\Delta u_i = U$$

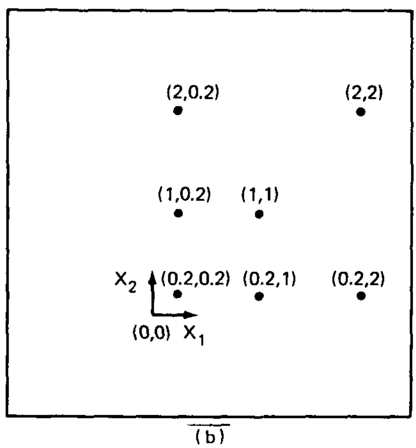

Figure 4 (b) Location notation for the calculated displacement fields.

2.00	46.0	53.5	62.8	71.7	78.9	83.7	85.9	85.7	83.4	79.7	74.9
1.80	54.6	63.3	73.9	83.9	91.6	96.4	97.8	96.3	92.5	87.0	80.6
1.60	65.7	75.4	87.3	98.5	106.8	111.1	111.3	107.9	101.8	94.2	85.8
1.40	80.3	90.7	103.7	115.9	124.5	128.0	126.2	120.0	111.0	100.5	89.8
1.20	99.9	110.0	123.6	136.7	145.3	147.2	142.2	132.1	119.1	105.2	92.0
1.00	126.6	135.0	147.7	161.0	169.4	168.6	158.6	142.6	124.5	106.9	91.1
0.80	163.8	167.7	176.8	189.3	197.0	191.6	173.4	149.2	125.1	103.8	86.1
0.60	216.2	212.6	212.9	221.8	228.5	214.3	182.2	147.2	117.2	93.8	75.9
0.40	289.3	277.7	262.9	259.1	264.4	230.8	174.7	129.0	97.4	75.7	60.4
0.20	386.0	374.2	349.8	313.3	308.6	211.8	129.6	88.7	66.0	51.8	42.3
					300	200	100				
	0.20	0.40	0.60	0.80	1.00	1.20	1.40	1.60	1.80	2.00	2.20

Figure 5 Displacement contours (displacement/slip on fault, mm/m) at the surface for vertical strike-slip fault. U_4 component of displacement; fault semilength, 1.0; depth to top of fault, 0.0; depth to bottom of fault, 2.0.

Theory and Computations. Steketee (1958) derived the general solution for the static displacement field in a semi-infinite medium using a Green's function approach. An analytic solution is possible, considering the rectangular coordinate system depicted in Fig. 4(a) and given certain assumptions (Chinnery, 1961). Chinnery simplified his expression by setting the depth parameter X_3 equal to zero. For this study the depth dependence was retained.

An example of the output is given in Fig. 5. To generalize the results as much as possible, we normalized the fault parameters by the half-length (L) of the fault. The displacements in the medium are normalized by the constant slip (Ω) on the fault and are in units of millimeters of displacement per meter of slip on the fault. The output has the following format:

- At each specified depth, the U_1, U_2 components are the horizontal displacement in the X_1, X_2 directions.
- The U_3 component is the vertical displacement (positive down).
- The U_4 component is the total displacement (i.e., the vector sum of U_1, U_2, and U_3).
- Only one quadrant is necessary because of symmetry.

For the actual computations, three different geometries were used for each of the strike-slip fault and dip-slip fault cases. Case 1 modeled long, shallow faults by setting D1 = 0.0 and D2 = 0.1. Case 2 modeled square, shallow faults by setting D1 = 0.0 and D2 = 2.0. Case 3 modeled square, deep faults by setting D1 = 2.0 and D2 = 4.0. In each case displacements were calculated at depth values of 0.0, 0.1, 0.5, 1.0, 2.0, 3.0, and 4.0 (recall that these numbers are normalized by the fault half-length). Table 2 gives an example of the calculations for one of the fault types. Because the amount of output for even the few cases considered was

Table 2 Displacement as a Function of Distance and Depth for a Long, Shallow Fault*

z	(0.2, 0.2)	(0.2, 1)	(0.2, 2)	(1, 0.2)	(1, 1)	(2, 0.2)	(2, 2)
0	140.5	104.9	6.0	16.5	18.2	4.3	5.3
0.1	117.6	88.8	5.7	15.8	18.6	4.2	5.5
0.5	16.6	16.9	4.2	11.1	15.5	4.2	5.7
1.0	2.8	4.4	2.5	5.5	9.0	3.7	5.2
2.0	0.3	0.8	0.8	1.4	2.6	2.2	3.1
3.0	0.1	0.2	0.3	0.5	0.9	1.1	1.7
4.0	0.0	0.1	0.2	0.2	0.4	0.6	0.9

**Displacement per slip on fault (mm/m) for a vertical strike-slip fault: fault semilength, 1.0; depth to top of fault, 0.0; depth to bottom of fault, 0.10.*

enormous, the results were tabulated and plotted in the following manner. For a specific surface location (X_1, X_2), the total displacement as a function of depth was plotted for each of the three cases for both strike-slip and dip-slip faults; a total of six curves for each plot was produced. We chose the seven locations shown in Fig. 4(b) as representative of the displacement field. In this manner, a large amount of information could be presented in a small number of figures.

Displacements as a Function of Depth. Details of the displacement pattern around faults vary greatly depending on the particular component of displacement and the type of fault (strike-slip or dip-slip). For the purpose of this report, only the total magnitude of displacement will be considered in detail. Because the total magnitude in effect averages all three components, the variations of displacement as a function of azimuth from the center of the fault are smoothed out. Therefore the variations in displacement that occur away from the fault at an angle of 45° are representative of the azimuth range (0 to 90°). Figure 6 illustrates the total displacement as a function of distance from the fault at the surface and a depth of 1.0. All three cases for both dip-slip and strike-slip faults are plotted on each graph. For most cases, the displacement drops off rapidly away from the fault. There are several cases for which the maximum displacement occurs away from the fault. At the surface the square, deep strike-slip fault reaches a maximum at a distance slightly greater than half a fault length away; the square, deep dip-slip fault reaches a maximum nearly a quarter of a fault length away. At a depth of half a fault length, the displacement curve for the long, shallow strike-slip fault changes drastically from the surface curve.

Displacements as a function of depth are presented graphically in Figs. 7 and 8. At a point near the fault (Fig. 5), all the curves except one have a maximum value of about 400 mm/m at the depth where the fault is located. For these cases, the displacements become very small within half a fault length away from the depth at which the maximum occurs. The one exception to these generalizations is the shallow, square fault. At a full fault length away, the displacements are diminishing but are still near the maximum value of the other cases.

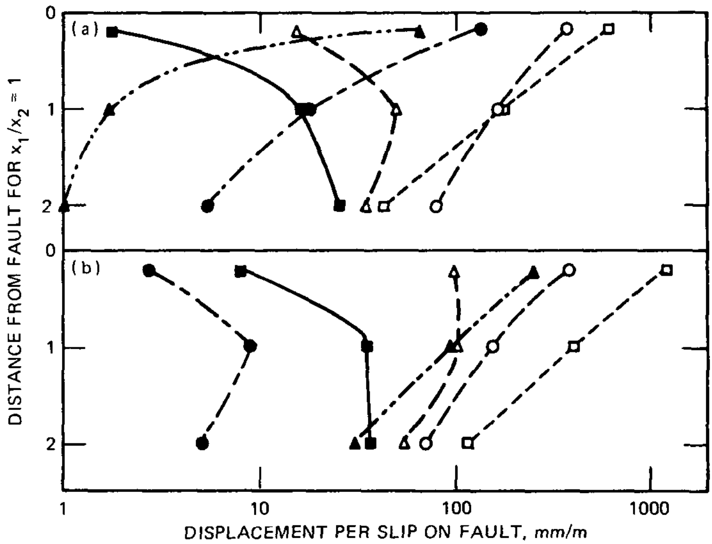

Figure 6 Displacement as a function of distance from fault at the surface and at a depth of one fault length. U_4 component of displacement. (a) Surface. (b) Depth of 1.0.

Vertical strike-slip faults:
● Case 1 (1) long, shallow fault
○ Case 2 (2) square, shallow fault
■ Case 3 (3) square, deep fault

Vertical dip-slip faults:
▲Case 4 (1) long, shallow fault
□ Case 5 (2) square, shallow fault
△Case 6 (3) square, deep fault

At point (1,1) the displacement curves tend to vary less drastically with depth (Fig. 8). With two exceptions (the long, shallow faults), the curves remain near a value of 100 mm/m. The strike-slip fault has displacements that increase rapidly from a minimum value at the surface to a maximum near a depth of 1.0. Dip-slip fault displacements decrease from a maximum at the surface to nearly zero at a depth of 4.0.

At point (2,2) the displacement curves have even less character. Almost all the curves are nearly linear with depth, lying between values of 30 mm/m and 100 mm/m. The lone exception is the long, shallow strike-slip, which again decreases monotonically to zero at a depth of 4.0.

Earthquake Source Parameter Relationships. In the preceding section, the fault model parameters were normalized by the half-length of the fault. Normalization generalizes the results, but it makes interpretation in terms of actual earthquakes more difficult. To use the results of the preceding section, we must relate magnitude to two other fault parameters—fault length and average fault displacement. At the present time empirical relationships are probably the best. These are usually confined to large magnitude earthquakes, and, because the larger earthquakes are of primary interest, the deficiency of data on small earthquakes is not critical. Some of the earlier data on magnitude and displacements are shown in Fig. 9 (from Chinnery, 1961).

The surface horizontal displacements for the 1906 San Francisco earthquake agree with those

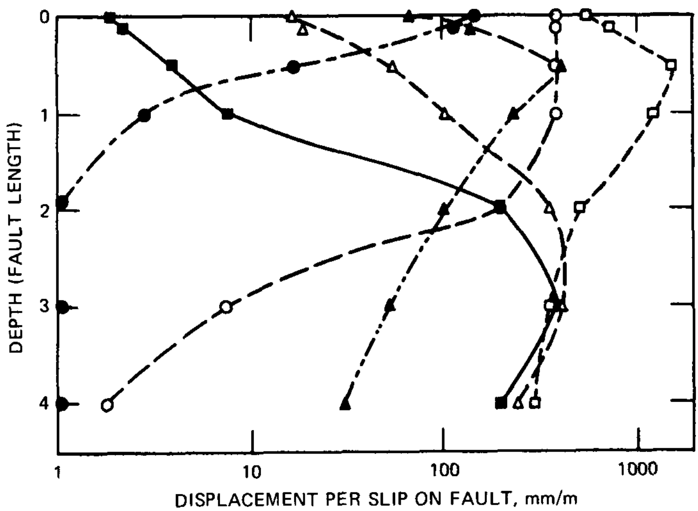

Figure 7 Displacement as a function of depth in terms of fault length at location (0.2, 0.2).

Vertical strike-slip faults:
● Case 1 (1) long, shallow fault
○ Case 2 (2) square, shallow fault
■ Case 3 (3) square, deep fault

Vertical dip-slip faults:
▲Case 4 (1) long, shallow fault
□ Case 5 (2) square, shallow faul
△Case 6 (3) square, deep fault

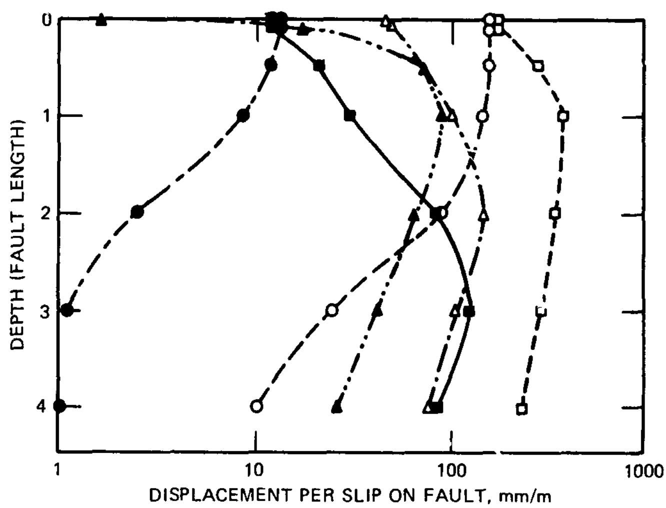

Figure 8 Displacement as a function of depth in terms of fault length at location (1,1).

Vertical strike-slip faults:
● Case 1 (1) long, shallow fault
○ Case 2 (2) square, shallow fault
■ Case 3 (3) square, deep fault

Vertical dip-slip faults:
▲Case 4 (1) long, shallow fault
□ Case 5 (2) square, shallow fault
△Case 6 (3) square, deep fault

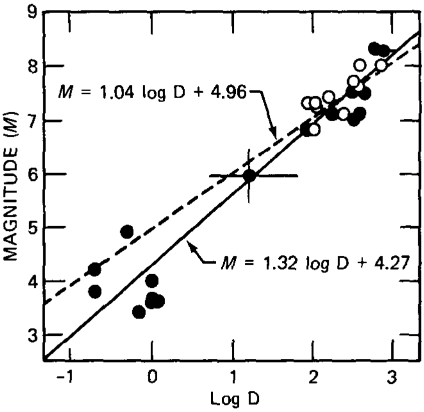

Figure 9 Magnitude plotted against the logarithm of displacement D. The solid line is a least-squares fit to all the points, and the dashed line is a least-squares fit at the points with M > 6.5.

calculated for a shallow, long, vertical strike-slip fault. The calculated displacements also agree with the magnitude displacement curve (Fig. 9).

Conclusions on Earthquake-Related Displacement Fields Near Underground Facilities. Analysis of observed "relative motion" data and calculations of displacement fields for various fault types and geometries indicate that

1. Most block motion displacements were recorded at the surface or at the free surface of tunnels.
2. Relative block motion can occur at depth, but displacement decreases markedly with distance and a decrease in energy source.
3. Analytical models have been developed to predict displacement as a function of distance and energy and fault motion as a function of distance and orientation from a given source.
4. Calculated displacement fields from vertical strike-slip and vertical dip-slip faults indicated that
   - Displacements drop off rapidly from the fault in most cases studied.
   - At a depth of one-half a fault length, the displacement curve for a shallow strike-slip fault (e.g., San Andreas) changes drastically from the surface displacement curve.
   - Of the models calculated, shallow, square, vertical strike-slip and dip-slip faults give the maximum displacement as a function of depth.

Numerical Simulations of Earthquake Effects on Tunnels for Generic Nuclear Waste Repositories

Introduction. Although damage in the ordinary sense of the word (i.e., roof falls, broken machinery, broken pipelines, etc.) will be of interest during the operational phase of a nuclear waste repository, in the long term greater interest will be in seismic damage that causes cracks that may enhance the permeability and the potential movement of groundwater.

A generic study was performed using numerical modeling techniques to determine under what conditions seismic waves generated by an earthquake might cause instability to an underground opening or create fracturing and joint movement that would lead to an increase in the permeability of the rock mass (Wahi et al., 1980). More specifically, the structural response of underground openings subjected to seismic motion was simulated, and a prediction was made of whether such loading might:

1. Cause collapse or other structural failure of these openings
2. Create new fractures or affect the existing fractures in the rock mass surrounding a repository in such a way as to increase permeability and, hence, groundwater circulation

The techniques had to model the system for all time ranges of interest, i.e., (1) very short range, for fully dynamic near-field response of underground openings to earthquake loadings (of the order of a few seconds); (2) medium to long range, for quasi-dynamic thermomechanical response during the retrievability phase (of the order of a few years); and (3) very long range, for quasistatic far-field response for thousands of years.

Inherent in the scope of the study were developments of realistic, nonlinear material models (thermoelastic), as well as failure models (plasticity, tensile fracture, and joint slip). The two-dimensional version of the STEALTH* codes formed the framework in which the problem-specific data were used to analyze a given repository system.

Three different rock types (salt, granite, and shale) were considered as host media for a single-level repository located at a depth of 600 m. The sensitivity analysis included variations in the in situ stress ratio, joint geometry, and pore pressures and the presence or absence of a shear zone. Data from three different events—the Parkfield, an Oroville aftershock, and an East Rand Proprietary Mines (ERPM), South Africa, mine tremor—were used to excite the rock mass. These data are summarized in Table 3. The loadings were considered representative of a "nearly large, a medium, and a small" earthquake to capture a broad range of realistic, seismic data. The material model description of each of the three rock types was unique in that each model displayed the characteristic response mechanisms of that rock.

For the purpose of analysis and comparison, criteria were defined for *Failure*, *Damage*, and *Permeability Enhancement.* Although the numerical limits used were somewhat arbitrary, they do represent meaningful physical thresholds. *Failure*, which was the primary criterion, was defined by a relative displacement of the tunnel walls, roof, or floor. A value of 0.05 (or greater) for the ratio displacement/tunnel dimension represented failure. Localized or regional *Damage* was considered to have taken place if new fractures, slippage along joints, or opening of existing joints occurred in excess of arbitrarily defined limits. The *Permeability Enhancement* criterion placed limits on the extent of crack opening and the extension of cracks beyond two tunnel diameters. In realistic terms, if permeability enhancement were limited to two tunnel diameters, the tunnel could be considered safe as far as groundwater flow enhancement is concerned.

All calculations performed were two-dimensional with plane-strain symmetry. Figure 10 shows the calculational grid, and Table 4 shows the initial specifications common to all rock types. All calculations were performed assuming homogeneous rock of the type specified; i.e., there were no layers of different rock to cause seismic reflection. The left and right boundaries were considered adiabatic; this implies a 52-m spacing of heat sources in the horizontal x direction. The mechanical boundary conditions were chosen to represent a single tunnel in order to model the asymmetric wave propagation of the earthquake. The matrix of calculations performed for salt is given in Table 5. The variations considered were the presence or absence of heat, the presence or absence of a shear zone, and three different earthquake motions. The matrix of calculations for granite is shown in Table 6 (the joint pattern used for granite is shown in Fig. 11). Three different pore-pressure distributions, five different in situ stress states, and two different earthquake motions formed the granite calculational matrix. The initial calculational matrix for shale consisted of two

Solids and thermal-hydraulics code for the Electric Power Research Institute, adapted from Lagrange TOODY and HEMP.

Table 3 Summary of Selected Acceleration Histories

Event	Date	Local magnitude	Focal depth, km	Recording site	Distance, km	Component directions	Peak acceleration, *g*	Peak velocity, cm/s	Peak displacement, cm
Parkfield earthquake	June 27, 1966	5.6	8.6	Temblor	12 (fault)	S25°W	0.35	22.5	5.5
						N65°W	0.27	14.5	4.7
						Vertical	0.13	4.9	1.4
Oroville aftershock	Aug. 11, 1975	4.3	2.6	CDMG-6	5 (epicenter)	N35°E	0.29	7.7	0.7
						S55°E	0.41	12.5	0.8
						Vertical	0.23	5.9	0.3
Mine tremor	Apr. 21, 1978	1.45§	2–3	ERPM	0.43	Longitudinal	0.95	1.47*	0.03
						Transverse	0.93	0.97*	0.02
						Vertical	0.85	1.23*	0.02

**Subsequent to this analysis the velocity used for the ERPM motion was found to be entered in meters per second rather than centimeters per second. Thus the motion actually modeled was more nearly equivalent to an earthquake of magnitude 3.7 than to the 1.45, which was recorded.*

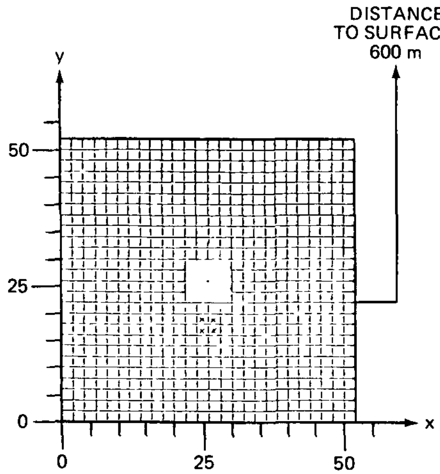

Figure 10 Calculational grid used for all runs except model with shear zone. Tunnel (or room) dimensions are 8 by 8 m. The bottom of the tunnel is 600 m below surface; i.e., the top of grid is 570 m below surface. Blocks marked with X are zones simulating canister heating.

Table 4 Assumptions Common to All Rock Types

Tunnel depth	600 m
Overburden stress	1 psi/ft
Natural temperature	35°C
Waste	10-yr-old spent fuel
Thermal loading	5-kW canisters 6.5 m apart in tunnels 52 m apart, 60 kW/acre
Sequence	0 yr, excavation; 0.5 yr, waste emplacement, heating; 5.0 yr, earthquake
No backfill, no ventilation	

cases; the first is shown in Table 7 and later was expanded to that shown in Table 8. (The initial joint pattern for shale is shown in Fig. 12.) This was done to include a weak shale (see Cases 1, 2, 5, and 6, rock mass properties), an "intact" shale (Case 3, intact properties), a strong shale-like rock (Case 4, isotropic mudstone), a smaller tunnel in shale (Case 6), and a reduced thermal loading (Case 5).

Instantaneous excavation of the opening was modeled in every case, and a period of 6 months was allowed after excavation before the placement of the heat-generating waste (Table 4). A heating phase of 4.5 yr (i.e., a total of 5 yr after excavation) was simulated in the first of a three-step calculational process for each case. After the heating phase, a stress-equilibration phase was computed so that the velocity field was very nearly zero before seismic activity. These first two steps of each simulation used constant-stress boundary conditions. In the third step, the seismic motion phase, boundary conditions were governed by the prescribed velocity histories of two components of the earthquake data. Isothermal boundary conditions were used at the top and bottom of the mesh.

The thermal response in each calculation for a given rock type (and the 60 kW/acre thermal load density) was essentially the same. As expected, salt showed the quickest dissipation of heat and shale, the slowest. The implications on thermal stresses are that a low heat dissipation rate results in higher thermal stresses for intact rock. Thermal loading does not necessarily alter the stability potential of a repository, however, as will be seen later.

Of all the salt cases considered, only one, that using the ERPM earthquake record, indicated unstable response. One of the acceleration records for the ERPM earthquake is shown in Fig. 13. For comparison, an acceleration record of the Parkfield earthquake is shown in Fig. 14. Note the significantly different frequencies, as well as the amplitude of the acceleration. In fact, by direct calculations in granite and by inference in

Table 5 Results for Salt Compared with Defined Criteria*

Property	Simulation Case 1	Case 2	Case 3	Case 4	Case 5
Joint geometry	X	X	X	X	X
Pore pressures	X	X	X	X	X
In situ stresses	$\sigma_H = \sigma_V$	$\sigma_H = \sigma_V$	$\sigma_H = \sigma_V$	$\sigma_H = \sigma_V$	$\sigma_H = \sigma_V$
Shear zone	X	X	X	X	y
Thermal loading	y	y	y	X	y
Earthquake	Oroville (0.41 *g*)	Parkfield (0.35 *g*)	ERPM (0.95 *g*)	Oroville (0.41 *g*)	Oroville (0.41 *g*)
			Criteria		
Failure	0	0	Failed (earthquake phase)	0	0
Damage	Fracturing void strain	0	Fracturing void strain	0	0
Permeability	0	0	0	0	0

**Symbols are X, none; y, yes; and 0, does not exceed criterion.*

shale, all three rock types were unstable for an 8- by 8-m opening when subjected to the ERPM motion. Since the ERPM earthquake was the "smallest" of the three that were used in the present analysis, it was curious that it should be the one to cause the most damage.

Since no damage from the ERPM had occurred in the quartzite mine where ERPM was recorded, the numerical simulations must predict a similar response for that environment. Otherwise, questions could be raised about the proposed techniques. An ERPM simulation was performed with the appropriate quartzite material-property data and geometry and without a heat source since there was no heat source in the real case. The results showed complete stability with respect to the ERPM motion. The relatively large depth (3 km), and therefore, large in situ stresses, the absence of joints, high tensile and compressive strengths, and the tunnel geometry were all thought to result in a stable environment for the quartzite mine.

Conclusions of the Model Study. Conclusions drawn from experiments with the three rock types—salt, granite, and shale—are listed in this section.

Numerical Model. The STEALTH 2D code was modified to simulate earthquake propagation accurately through a two-dimensional mesh. Material models were developed to model the nonlinear behavior of salt, granite, and shale. A number of complex cases were modeled in a three-stage calculational process taking into account the effects of jointing, heating, water saturation, and different in situ stress combinations. The methodology is suitable for studying the effects of earthquake motions on underground structures.

Salt. Conclusions from the salt runs (Table 5) are

1. During the heating phase the creep of the salt helped to relax the stress concentrations around the tunnel.

2. The rock mass was able to withstand the Oroville and Parkfield earthquakes without significant frac-

Table 6 Results for Granite Compared with Defined Criteria*

Property	Simulation Case 1	Case 2	Case 3	Case 4	Case 5	Case 6	Case 7	Case 8
Joint geometry	G_1	G_1	G_1	G_1	G_1	G_1	G_1	G_1
Pore pressure	X	X	X	H1	H2	X	X	X
In situ stresses	$\sigma_H = \sigma_V$	$\sigma_H = \frac{1}{2}\sigma_V$	$\sigma_H = 2\sigma_V$	$\sigma_H = \sigma_V$	$\sigma_H = \sigma_V$	$\sigma_H = \sigma_V$	$\sigma_{H1} = 2\sigma_V$ $\sigma_{H2} = \sigma_V$	$\sigma_H = \frac{3}{2}\sigma_V$
Shear zone	X	X	X	X	X	X	X	X
Thermal loading	y	y	y	y	y	y	y	y
Earthquake	Oroville (0.41 *g*)	Oroville (0.41 *g*)	Oroville (0.41 *g*)	Oroville (0.41 *g*)	Oroville (0.41 *g*)	ERPM (0.95 *g*)	Oroville (0.41 *g*)	Oroville (0.41 *g*)
Criteria								
Failure	0	0	Failed (before earthquake phase)	0	0	Failed (earthquake phase)	Failed (before earthquake phase)	0
Damage	0	0	Slip	0	0	Failed (earthquake phase)	Slip	0
Permeability	0	0	Exceeded	0	0	Exceeded	Exceeded	0

**Symbols are G_1, joint pattern, as shown in Fig 11, H1, pore pressure from 0 at 4 m to hydrostatic at 30 m from tunnel center, H2, pore pressure from 0 at 4 m to hydrostatic at 12 m from tunnel center, σ_H, horizontal stress, σ_V, vertical stress, X, none, y, yes, and 0, does not exceed criterion.*

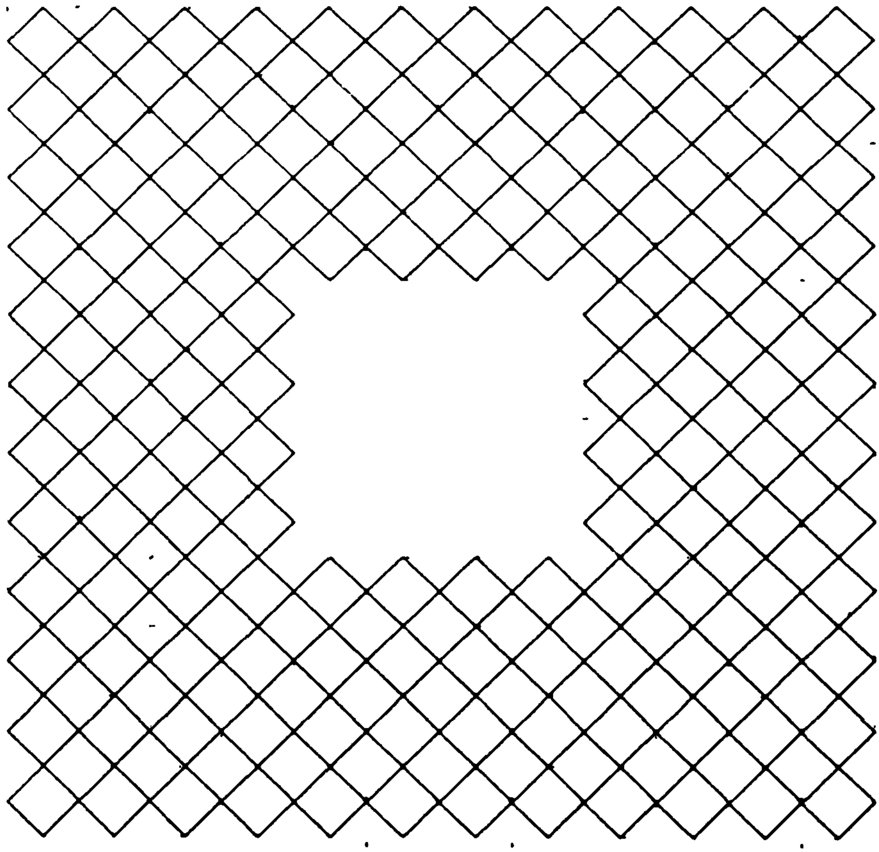

Figure 11 STEALTH mesh for granite (interior zones only) showing joint pattern (solid lines) and zone boundaries (dotted lines).

turing or instability around the tunnel.

3. The ERPM mine tremor was greatly amplified around the tunnel, and fracturing occurred as a result of the seismic loading. The results for salt are compared with the defined criteria in Table 5.

Granite. Conclusions drawn from the granite runs (Table 6) are:

1. During the heating phase some slip occurred along the joints and was accompanied by occasional cracking of intact rock in the immediate vicinity of the opening. There was no tendency for major through-going cracks to form.

2. No cracks were developed as a result of passing the Oroville motions through the rock mass under hydrostatic loading conditions, although some additional slippage along joints occurred in zones that had already experienced slip during the heating phase.

3. For certain high horizontal stress scenarios (e.g., when one or both of the in situ horizontal stresses were twice the overburden), major slippage occurred along the joints even before the earthquake exaggerated this slip. The inference is that the condition of high horizontal stress might pose problems for an under-

Table 7 Results for Shale Compared with Defined Criteria*

Property	Simulation Case 1	Simulation Case 2
Joint property	S_1	S_1
Pore pressure	X	H1
In situ stresses	$\sigma_H = \sigma_V$	$\sigma_H = \sigma_V$
Shear zone	X	X
Thermal loading	y	y
Earthquake	Oroville (0.41 *g*)	Oroville (0.41 *g*)
Anisotropy	y	y
Criteria		
Failure	Failed (earthquake phase)	Failed (heating phase)
Damage	Slip void strain	Slip void strain
Permeability	Exceeded	Exceeded

**Symbols are S_1, joint pattern, as shown in Fig. 12; H1, pore-pressure from 0 at 4 m to hydrostatic at 12 m; X, none; and y, yes.*

ground waste repository under the assumed jointing and loading conditions.

4. Although the assumed pore pressure distributions affect the stress distribution around the tunnel and the amount of joint slip, in granite they do not appear to have very much influence on the transmission of the earthquake motions.

5. The ERPM motion causes the hypothetical repository configuration to become unstable. Large-scale deformation resulting from cracking and slippage causes this failure. The results for granite are compared with the defined criteria in Table 6.

Shale. Conclusions drawn from the results of shale runs (Tables 7 and 8) are:

1. During the heating phase heat was trapped near the cavity (as a result of the lower thermal conductivity of shale in comparison with salt and granite); this leads to higher stress gradients around the tunnel.

2. For the cases in which a pore-pressure distribution existed, major slipping along joints occurred during the heating phase. Residual velocities following the heating phase but before the earthquake were of the order of the peak velocities that would be expected during the Oroville earthquake. This was done so that the earthquake effects would be small in comparison with the amount of slip driven by thermomechanical and in situ stresses. Large deformations at the tunnel occurred during the heating phase for the various weak shale cases.

3. Applying the Oroville motion to the weak shale case in which no pore pressure existed resulted in tunnel instability after 0.5 s of the earthquake motion. It is likely that the tunnel was only marginally stable before the

Table 8 Additional Results for Shale Compared with Defined Criteria*

Property	Simulation Case 3	Case 4	Case 5a	Case 5b	Case 6a	Case 6b
Joint geometry	S_1	S_2	S_1	S_1	S_1	S_1
Pore pressure	X	H1	X	H1	X	H1
In situ stresses ($\sigma_H = \sigma_V$)	y	y	y	y	y	y
Shear zone	X	X	X	X	X	X
Thermal loading (60 kW/acre)	y	y	15 kW/acre	15 kW/acre	y	y
Earthquake Oroville (0.41 *g*)	y	y	y	y	y	y
Anisotropy	y	X	y	y	y	y
Opening size	8 by 8 m	8 by 8 m	8 by 8 m	8 by 8 m	4 by 4 m	4 by 4 m
			Criteria			
Failure	Failed (earthquake phase)	0	Failed (earthquake phase)	Failed (heating phase)	Failed (earthquake phase)	Failed (heating phase)
Damage	Fracturing void strain	Exceeded	Slip void strain	Slip void strain	Exceeded	Slip void strain
Permeability	Exceeded	Exceeded	Exceeded	Exceeded	Exceeded	Exceeded

**Symbols are S_1, joint pattern, as shown in Fig. 12; S_2, isotropic mudstone, joint spacing 30 m; H1, pore pressure from 0 to 4 m to hydrostatic at 12 m; X, none; y, yes; and 0, does not exceed criterion.*

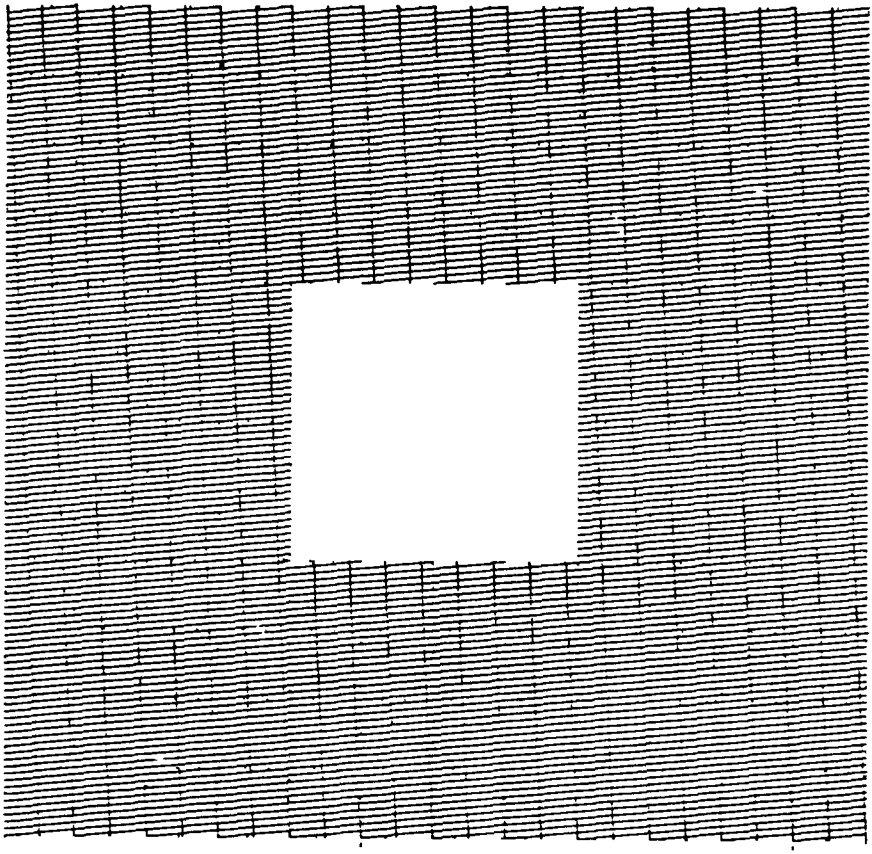

Figure 12 STEALTH mesh for shale (interior zones only) showing joint patterns (solid lines) and zone boundaries (dotted lines).

earthquake, since the Oroville motion did not cause failure in either a granite or a salt repository. Also, the fact that relative deformation at the tunnel walls at the end of the equilibrium phase in shale was many times greater than in granite indicates incipient instability.

4. Using material property data for intact shale resulted in a relatively more stable structure during the heating phase, as expected. It was not sufficiently strong to withstand the Oroville aftershock motion, however, and instability resulted during the early part of the motion. Since the tensile and compressive strength values were the same for the weak and the intact shale and since tensile failure and joint slip are the dominating failure mechanisms, instability with respect to the seismic motion is not totally unexpected.

5. The 8- by 8-m opening in isotropic mudstone remained intact with respect to the Oroville aftershock motion. The displacements at the end of the heating phase were substantially lower than any of the weak or intact shale cases. Although finite pore pressures were used, the joint spacing was much greater; this helped to stabilize the structure. A lack of anisotropy also contributed to its stability. Some new cracks, partial and through-going, did develop during the

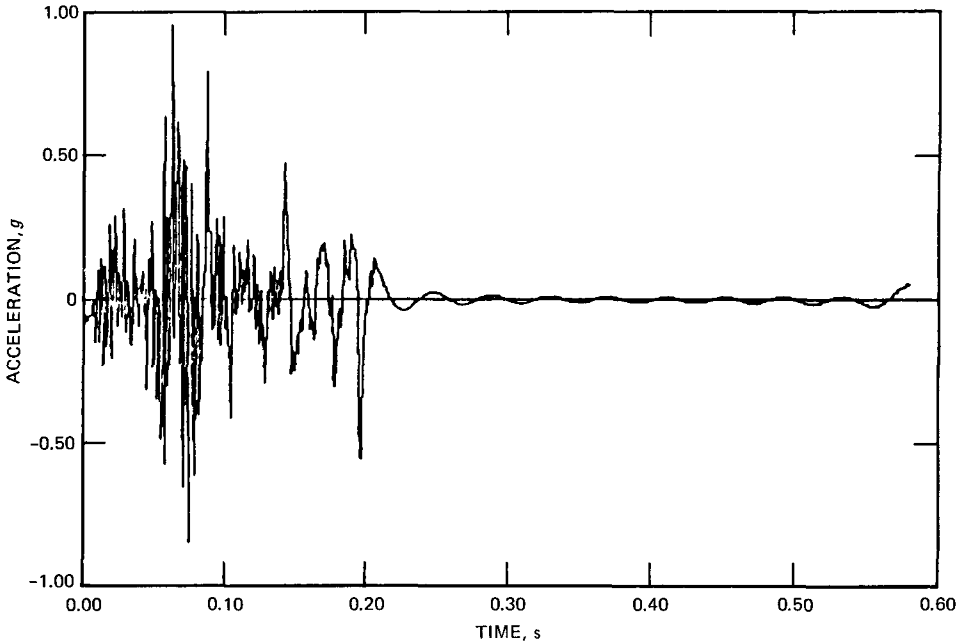

Figure 13 Acceleration record of the East Rand Proprietary mines, South Africa, 1978, longitudinal component.

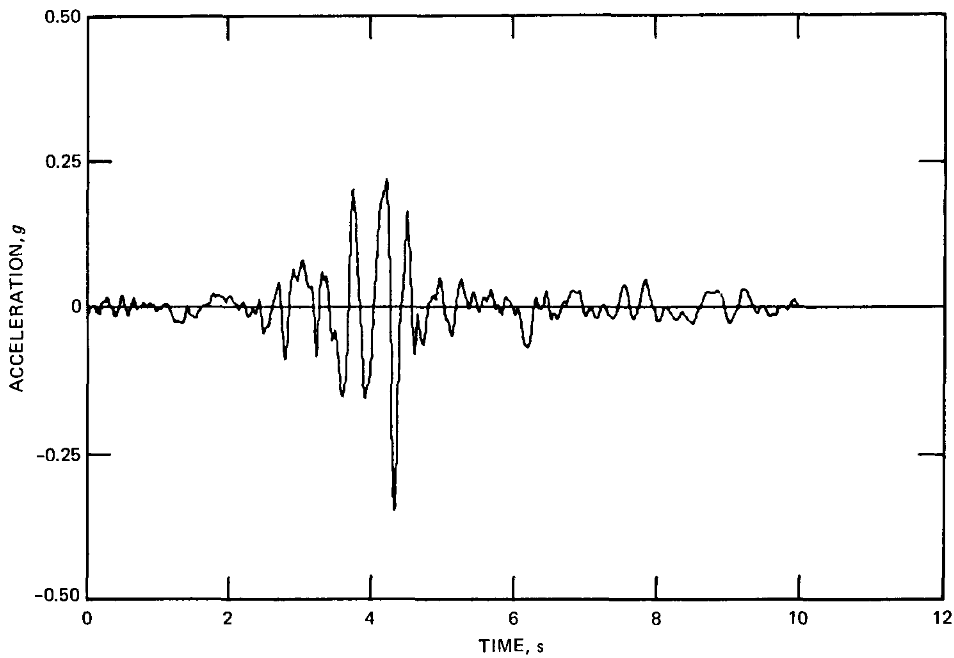

Figure 14 Acceleration record of Temblor, Parkfield, 1966, S25°W component.

heating phase. The new cracks were limited to two distinct regions—in the vicinity of the tunnel and in the central portion near the mesh bottom. No additional cracks or void strains were introduced by the passage of the Oroville aftershock.

6. For reduced thermal loading in weak shale (15 kW/acre, Case 5a), the response during heating, as well as seismic activity, was nearly identical to that of Case 1 (60 kW/acre). Since this may or may not be typical of other rock types, no general conclusions should be made about the influence of thermal load density.

7. As expected, a smaller tunnel (4 by 4 m) in weak shale resulted in a relatively more stable response during the heating phase. The tensile perturbations experienced during the seismic activity were strong enough to cause tunnel collapse of the 4- by 4-m opening, however. Also, the presence of pore water invariably resulted in unstable response in weak and intact shales and in large and small tunnels even before any seismic activity. The results for shale are evaluated against the defined criteria in Tables 7 and 8.

On the basis of the analyses performed thus far, generic shale appears to be an incompetent rock for an underground waste repository at the proposed depth of approximately 600 m. The empirical investigations that relate the size of an unsupported opening in a given rock type to its stand-up time (data based on actual case histories) support the general conclusions of the present study.

We should emphasize that this entire modeling study was generic in nature and that, for a specific site, rock type, and design, studies in greater depth need to be performed.

Conclusions

In general, mines and subsurface facilities sustain very much less damage than surface facilities during the same earthquake. This statement is supported by both empirical and analytical evidence.

Empirical evidence includes

1. The study by Rozen (1976) and Dowding (1978) showed "no damage" to shallow tunnels below surface accelerations of 0.2 *g*, which would have caused damage and partial collapse of masonry structures and the fall of chimneys, monuments, and towers. The same study showed only minor damage between 0.2 and 0.5 *g*, which would have caused destruction of masonry and frame structures, dams, dikes, and bridges.
2. The study of Duke and Leeds (1961) showed little damage to mines and tunnels outside the epicentral region. Even within the epicentral region, damage was confined to poor quality construction, portals, and parts of the facility that crossed a fault.
3. Numerous Japanese studies show surface-depth displacement ratios from 1.2 to 4.2.
4. Studies of the region of the Alaskan earthquake of 1964 showed only a few loose rocks in mines and tunnels, whereas surface damage was extreme.
5. Coal miners underground felt no effects of the 1960 Chilean earthquake, one of the strongest on record.
6. No damage occurred to mines and tunnels during the 1970 Peruvian earthquake.
7. The fact that there have been so few studies of subsurface effects of earthquakes indicates that earthquake damage has not been a significant problem to the operators of subsurface facilities. Most such studies are pursued for academic reasons or by some organization that is planning a subsurface facility rather than by current operators.

Analytical evidence is provided by the numerical model study. No failure and little damage was caused to a tunnel in salt or granite by either the

Oroville (0.41 g) or the Parkfield (0.35 g) earthquakes, except where the horizontal stress was twice the vertical stress. The accelerations of both of these earthquakes are greater than many surface structures are designed to withstand. In addition to supporting statements that subsurface damage is substantially less than surface damage, the studies also provided some insight into the potential for damage to subsurface facilities.

1. Shale tunnels appear to be more vulnerable to damage than granite or salt, although the massive mudstone did not fail. Neither the decreased heat load nor the smaller tunnel size appeared to have much effect.

2. In situ stress is an important parameter in determining tunnel damage. In a stress regime where the horizontal stress is as much as twice the vertical, damage occurred.

3. The damage brought about by the ERPM earthquake raises some questions. One interpretation is that high-frequency high-acceleration close-in earthquakes may cause damage. A similar acceleration at a similar frequency would not cause damage to surface structures because of their long natural periods. These wavelengths are more comparable to the tunnel dimensions, and, by the same token, the longer wavelengths that remain after earth filtering of the waves from large earthquakes at moderate distance do not do damage. This reasoning is compatible with the empirical observations. Thus only close-in earthquakes provide short enough wavelengths to cause damage to mines. Surface waves would be expected to have little effect on mines because they may not penetrate to depth or they are not of short enough wavelength. An alternative interpretation of the effects of the ERPM earthquake is that it is unrealistically modeled. Remember that the model is two dimensional. It is well known that, as higher frequency sensitive instruments are deployed, higher accelerations will be measured. These accelerations have little significance for surface structures. For subsurface facilities, the question of which of the interpretations is applicable remains.

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Design of Repository Sealing Systems—1981

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Isolating nuclear waste in geologic repositories will require the sealing of penetrations such as access shafts and tunnels, disposal rooms, and exploration boreholes. This paper discusses seal designs developed for a repository in bedded salt referenced to the stratigraphy of southeastern New Mexico. Designs are based on a multiple component concept whereby individual components are designed for a specific function and location. For a repository in salt the major function of the seals is to exclude groundwater inflow. Two main types of component are included for this purpose: (1) Bulkheads are dense concrete structures keyed into the walls of the penetration and are intended to reduce flow at the interface between the seal and the salt. (2) Backfills are granular materials compacted in place in the penetration. In the repository the major backfill material is crushed salt, which is expected to consolidate and recrystallize as the rooms close in response to salt creep. Densely compacted clays will be used as backfill in the shafts closer to potential sources of water inflow. 22 ref. 10 fig., 1 tab.

Introduction

Isolating nuclear waste in deep, mined repositories will require the sealing of all penetrations into or near the facility. Types of penetrations (Fig. 1) include exploratory boreholes, shafts, and entry drifts (tunnels). In addition, backfilling and internal separation of the waste disposal chambers or "storage rooms" are considered to be part of the overall seal system. The design of seals is an integral part of the Department of Energy National Waste Terminal Storage (NWTS) program for the permanent isolation of commercial and defense waste. The design program is being conducted by D'Appolonia Consulting Engineers for the Office of Nuclear Waste Isolation (ONWI), with major support from a number of field and laboratory contractors.

The program began in 1978 with reviews of work by previous investigators and of sealing techniques used for petroleum production and storage, mining, civil construction, deep-well disposal, and military operations (Office of Nuclear Waste Isolation, 1978, 1979). Conclusions were that significant materials and installation experience does exist, but the bulk of this knowledge for conventional seals is of only partial value for repository sealing. Major differences center around design functions, quality control and licensing procedures, and, most importantly, longevity requirements for repository seals.

Subsequent completed studies defined a design approach for repository seals (Office of Nuclear Waste Isolation, 1980a), evaluated research requirements for seal materials

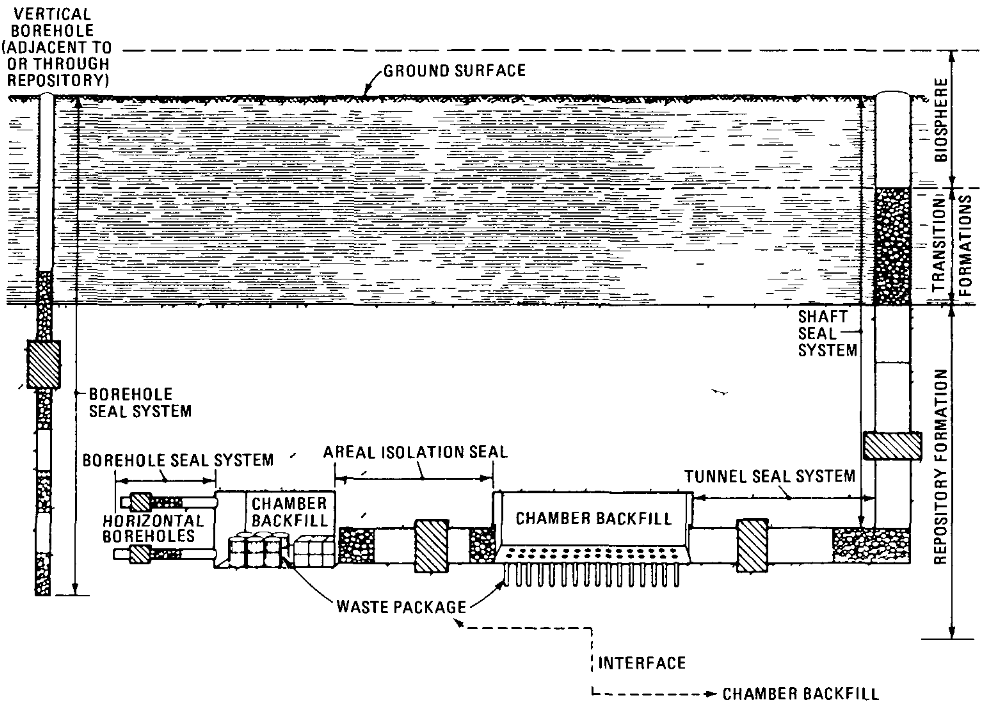

Figure 1 Penetration sealing applications. Types of seals are borehole, horizontal and vertical; shaft; tunnel, including areal isolation seals; and chamber backfill.

(Office of Nuclear Waste Isolation, 1980b), and compared the requirements for sealing shafts, tunnels, and boreholes (Office of Nuclear Waste Isolation, 1981a). An important ongoing activity has been close coordination of design with supporting field and laboratory test programs. Concurrent with the ONWI design program, preconceptual designs for penetration seals for a repository in basalt have been developed by Rockwell Hanford operations and Woodward-Clyde Consultants (Taylor et al., 1980).

Current design activities focus on developing conceptual designs for a repository in bedded salt using the Waste Isolation Pilot Plant (WIPP) site at Los Medanos in southeastern New Mexico as a reference case. This paper describes a general design approach for penetration seals which can be applied to any candidate repository site, with proper regard for site-specific geologic and design conditions. A detailed description of the interim designs developed for bedded salt is given as an initial example.

Seal Zone Components

Figure 2 shows that the seal zone consists of three potential pathways for fluid flow—the seal itself, the interface between the seal and the host rock, and a disturbed zone that may exist in the host rock. The disturbed zone is assumed to exist as a result of stress relief or damage by the excavation process or by weathering. Seal design must address all three components of the seal zone.

Interface Studies. Direct evidence regarding the importance of the

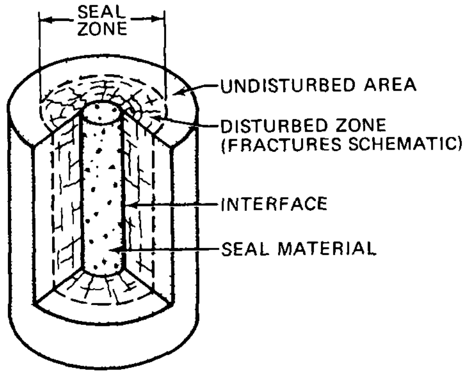

Figure 2 Seal zone components.

seal-host-rock interface has been obtained from field and laboratory tests on borehole plugs. The Bell Canyon test, conducted by Sandia National Laboratories and Systems, Science and Software at the WIPP site, involved placing a 2-m-long cement plug in a borehole 20 cm in diameter (Christensen and Peterson, 1981). The plug was set in an anhydrite formation at a depth of 1368 m, about 11.6 m above the Bell Canyon formation aquifer (Fig. 3). A bridge plug and an instrument package, including an automatic tracer release, were placed in the borehole below the plug. Flow tests through the cement plug were conducted by deflating the bridge plug, thus exposing the cement plug to water from the Bell Canyon aquifer at approximately 12.4 MPa pressure. A packer was set above the plug to monitor fluid buildup, shut-in pressure, and tracer arrival time. The measured flow above the cement plug was approximately 0.5 cm^3/min, and the flow velocity, estimated from tracer arrival time, was 1.2 m/d.

Estimating plug permeability requires knowledge of the cross-sectional area of flow. If we arbitrarily take this area to be twice that of the well bore (to include the disturbed zone), the equivalent permeability is approximately 50 μdarcies. This compares with values measured in the laboratory for both the cement and the intact anhydrite (small samples) in the range 0.1 to 1.0 μdarcy. Since there are no penetrations through the plug, the major contribution to flow appears to be in the disturbed zone or, more probably, at the interface. Full details of test procedures, including grout development, are given by Christensen and Peterson (1981).

Laboratory tests conducted at Terra Tek, Inc., Salt Lake City, provided information to make a valuable comparison with the Bell Canyon test. A cement plug 48 cm long and 20 cm in diameter was placed in a borehole drilled in the laboratory into a block of anhydrite (Pratt et al., 1980). The borehole was drilled and the plug placed in a custom-built pressure vessel, which allowed a nearly full-scale simulation of in situ conditions, e.g., overburden stress, bit pressure, and mud pressure. In this case the test parameters were set to approximate the conditions of the in situ Bell Canyon test. After the plug was cured, a flow test was performed at a differential fluid pressure of 13.8 MPa. Dye was introduced in the permeant to highlight flow paths and the extent of penetration into any disturbed zone. After the flow test was completed, the sample was removed from the pressure vessel and sectioned for examination of the interface and disturbed zone. Preliminary results show a steady-state flow of 5.0 cm^3/min (Terra Tek, Inc., 1981). If we equate the test conditions with those of the Bell Canyon test and assume a flow area of twice the cement plug area, the equivalent permeability is 80 μdarcies, 1.6 times that for the Bell Canyon test.

Examination of the sectioned plug and the dye penetration revealed clearly that the flow had been concen-

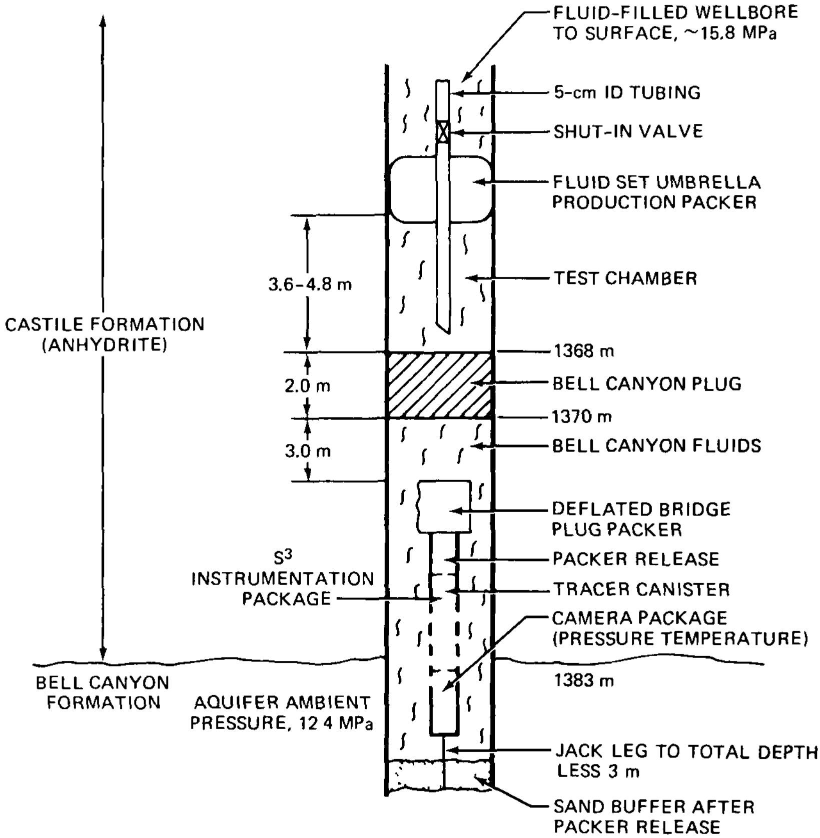

Figure 3 Bell Canyon test configuration, not to scale. (Data from Christensen and Peterson, 1981.)

trated at the interface in a much smaller area than that arbitrarily assigned for calculating equivalent permeability. This test provides strong evidence that, at least for borehole plugs in anhydrite, the interface is a more likely source of leakage than the disturbed zone. Additional laboratory tests are planned; plugs composed of different materials will be set and tested in a variety of host-rock types.

Disturbed Zone Studies. Field tests to evaluate the extent and nature of the disturbed zone around a tunnel in granitic rock are being conducted at the Colorado School of Mines test mine in Idaho Springs, CO (Hustrulid et al., 1980). The tests are conducted in an experimental room 5 m wide by 3 m high excavated in sections by different controlled blasting methods. Tests include air-permeability and geophysical measurements in a pattern of boreholes drilled radially from the tunnel. Preliminary results indicate that the permeability may be increased near the tunnel wall, but this trend is difficult to confirm because of the variability in the jointed rock. If a distributed zone does exist, it appears to be limited to an area less than 1 m wide (Office of Nuclear Waste Isolation, 1980c).

Some evidence regarding the disturbed zone has also been obtained from the macropermeability tests conducted at the Stripa Mine in Sweden (Office of Nuclear Waste Isolation, 1980d). In a test involving monitoring the total water inflow in a closed-off section of a tunnel in fractured, granitic rock, hydraulic pressures were monitored at intervals away from the tunnel walls in radial boreholes. The pressure distributions obtained indicate that the radial permeability is reduced in a zone approximately 2 to 3 m wide surrounding the tunnel, presumably as a result of stress concentration. The test provided no evidence regarding the effects of disturbance on permeability parallel to the tunnel.

In the general mining and civil engineering fields, there have been many investigations of the disturbance produced by excavation, but these have not included permeability measurements. Therefore, with the exception of the results of the Colorado School of Mines and the Stripa tests, very little information is available on the permeability of the disturbed zone and how that zone might be affected by the excavation method used. Analyses of the disturbed zone which relate predicted stress distributions to possible effects on permeability are being conducted. Also, future field testing, either at candidate repository sites or generic test facilities, will include detailed characterization of the disturbed zone. Valuable evidence regarding disturbed zone characteristics for boreholes will be obtained from the laboratory tests at Terra Tek, Inc., described previously.

Design Goals for Seals

The basic approaches that guide the design program were developed in 1979 and were published by the Office of Nuclear Waste Isolation (1980a). The primary goals are to develop safe seal designs that can be constructed at reasonable costs and to obtain licensing for seals within the overall NWTS program. Resulting technical requirements are to: (1) characterize penetration, environmental, and loading conditions that will be experienced; (2) develop suitable seal materials and geometries considering specific functions, installation procedures, and costs; and (3) develop techniques to verify the adequacy of designs.

The Office of Nuclear Waste Isolation (1980a) examined alternative goals or criteria that could be used eventually to quantify seal requirements. The recommended design goal related to radionuclide release is that *the radioactive migration rate through the seal zone should always be less by a specified factor of safety than an acceptable level, which is determined by the consequence of such a release.* The factor of safety would be established on a case-by-case basis, considering institutional standards and uncertainties in the design. The natural decay of the stored radioactive materials permits the design criteria and factor of safety to be functions of time.

Functional Criteria and Design Bases. The basic design goal of limiting radionuclide release below an acceptable level will be achieved by evaluating the processes by which radionuclides could be released and carried to the biosphere and then establishing the functions required of seals to limit releases.

Transport by groundwater is the recognized mode by which radionuclides may be released from a repository. Hence, depending on site conditions, the most important functions for seals may be to limit groundwater ingress to the repository and, assuming water enters the repository, to limit the quantity and travel time of flow from the repository to the biosphere. This function will be particularly important for a repository in salt where it is essential to limit dissolution of the seals and adjacent host rock and where saline groundwater could lead to rapid degradation of the waste package. This function will be less important in a more permeable or less soluble host rock where there will be a significant inflow from the host rock itself.

The necessity to limit groundwater inflow is a functional criterion for seals. The quantitative limiting value below which the flow must be maintained is established as a design basis. Depending on specific site conditions, design bases might be established for such parameters as radionuclide migration (quantity against time), volume of groundwater ingress, groundwater travel time to the biosphere, and room deformation (which, if excessive, could disrupt overlying aquifers and create pathways to the biosphere). Design bases will be established by the designer from consequence analyses, taking into account regulatory requirements. Work to establish quantitative design bases for specific sites is in progress.

Regulatory Requirements for Penetration Seals. The standards for repository performance will be established by the Environmental Protection Agency (EPA). A preliminary standard (Environmental Protection Agency, 1981) defined the maximum allowable release rates in terms of inidividual isotopes, referenced to the quantity of waste in the repository. The standard would apply for a period of 10,000 yr after disposal. A relaxed standard would be accepted for very unlikely release scenarios. The preliminary standard is derived from consideration of acceptable health effects and is, thus, in the same form as the recommended design goal for seals.

The Nuclear Regulatory Commission (NRC) will license the construction, operation, and decommissioning (including sealing) of repositories for commercial wastes. In the proposed rulemaking for high-level waste disposal, published as 10 CFR Part 60 (Nuclear Regulatory Commission, 1981), NRC proposed a number of performance criteria for repository subsystems that are to be applied in addition to the EPA standard for the repository as a whole. These include standards for containment in the waste package, maximum release rates from the engineered system, and minimum groundwater travel time from the engineered system to the accessible environment (the biosphere). In addition, NRC proposed a standard for repository seals: "sealed shafts and boreholes shall inhibit transport of radionuclides to at least the same degree as the undisturbed units of rock through which the shafts or boreholes pass."

The regulations are not final, and the suggested standard for seals is still under discussion. The stated DOE position (Department of Energy, 1980) is that: ". . . penetration sealing must provide a barrier with sufficient integrity to ensure acceptable consequences. Hence, only adequacy is required, and it should be determined on a site-specific basis." This is in accordance with the ONWI design goal that the standard should be related to an acceptable limit as determined by consequence analysis. In practice, on a site-specific basis, acceptable release from penetration seals must be related to acceptable release from the repository as a whole. Hence, the performance of seals must be compared with that of the host rock, and it follows that the requirements for seals will vary from site to site according to the permeability and sorptivity of the host rock. Requiring that seal performance be equivalent to that of undisturbed host rock is unnecessarily restrictive, however. Table 1 summarizes regulatory criteria or guidelines that may apply to penetration seals.

Design Approach

Design Stages. Designs are developed using an iterative approach incorporating characteristics of site-specific host rock and properties of seal materials as information becomes available from concurrent field and laboratory testing. Three specific design stages are identified:

1. A schematic design is a media-specific working design that incorporates available information on host media, potential host environments, seal materials, and modeling. It is not considered a prototypical seal design. As a working design, it evolves as new data and information become available; thus schematic sealing designs for all media are updated annually. The schematic design serves three significant purposes: It summarizes progress in seal design development; it aids in developing media-specific

Table 1 Summary of Proposed Criteria and Guidelines for Penetration Seals

Characteristic	Requirement	Reference
Containment in waste package	1,000 yr	Nuclear Regulatory Commission, 1981
Release from waste package or repository backfill	$<10^{-5}$ of inventory per year (for each nuclide)	Nuclear Regulatory Commission 1981
Release to accessible environment	Release limits specified for individual isotopes (Ci, cumulative over 10,000 yr)	Environmental Protection Agency, 1981
Groundwater travel time to accessible environment	1,000 yr	Nuclear Regulatory Commission, 1981
Isolation period	10,000 yr	Nuclear Regulatory Commission, 1981; Environmental Protection Agency, 1981
Seal performance	Inhibits transport of radionuclides to at least the same degree as the undisturbed rock	Nuclear Regulatory Commission, 1981
	Total allowable releases from repository as determined by EPA	Office of Nuclear Waste Isolation, 1981b
	Releases less than an acceptable limit as determined by consequence analysis	Office of Nuclear Waste Isolation, 1981a

design considerations; and it assists in pointing out data deficiencies and research needs that must be addressed before conceptual designs are formulated.

2. A conceptual design is a site-specific design that is almost complete, except for minor problems that have been identified and for which a solution or a design mitigation appears imminent. The conceptual design establishes a detailed program to resolve all issues by continuing field and laboratory testing. Conceptual designs are reviewed by a wide section of the technical community.

3. A preliminary design is a site-specific design that is technically acceptable and can be substantially demonstrated as being appropriate to technical and nontechnical communities. It must be suitable for submittal as part of the application for a repository construction license. Final designs are based on actual conditions encountered in the repository.

Figure 4 illustrates the relationships among the various design stages and concurrent laboratory and field testing. As shown, the design process evolved from an initial evaluation of the state of the art of shaft and borehole sealing (Office of Nuclear Waste Isolation, 1979) and from two reports that evaluated the broad objectives for laboratory and field studies (Office of Nuclear Waste Isolation, 1979, 1980b). A similar design procedure, from schematic to preliminary designs, will be followed for each site considered and will involve a progressive review of working designs, taking into account specific repository information as it becomes available. The procedure will incorporate results from performance assessment (mainly modeling) and will integrate laboratory and field test results.

The design approach recognizes the necessity for site-specific designs to focus on actual site problems and to avoid considering unrealistic conditions or combinations of conditions that might be assigned to a generic site. The first schematic and conceptual designs are being developed for bedded salt. They are referenced to site-specific conditions at the WIPP site in New Mexico, which at the present time has been explored in greater detail than any other candidate repository site. Although the WIPP repository is intended for transuranic waste, the designs are being developed so that they could be applied without major modification to a different repository layout designed specifically for high-level waste or spent fuel. Thermal loadings used in modeling the seal designs are based on conditions for high-level waste or spent fuel.

Schematic and conceptual designs for other candidate repository sites will be developed between 1982 and 1985. With regard to designs for basalt, significant benefits will be gained from the preconceptual designs developed by Rockwell Hanford. Schedules for specific sites are linked to site characterization schedules so that the designs are based on site-specific data. Preliminary designs for one or more sites selected for license application will be completed by fiscal year 1987.

Design Support Activities. An important element of the design approach is the integration of office, laboratory, and field studies. Costs and schedules usually increase as work progresses from analyses to laboratory studies to field investigations. At the same time, however, certain factors (such as seal-host-rock interface characteristics and suitability of an installation technique) can be determined only in the laboratory or the field. Analytical studies must be used to plan and supplement the limited numbers of tests by perform-

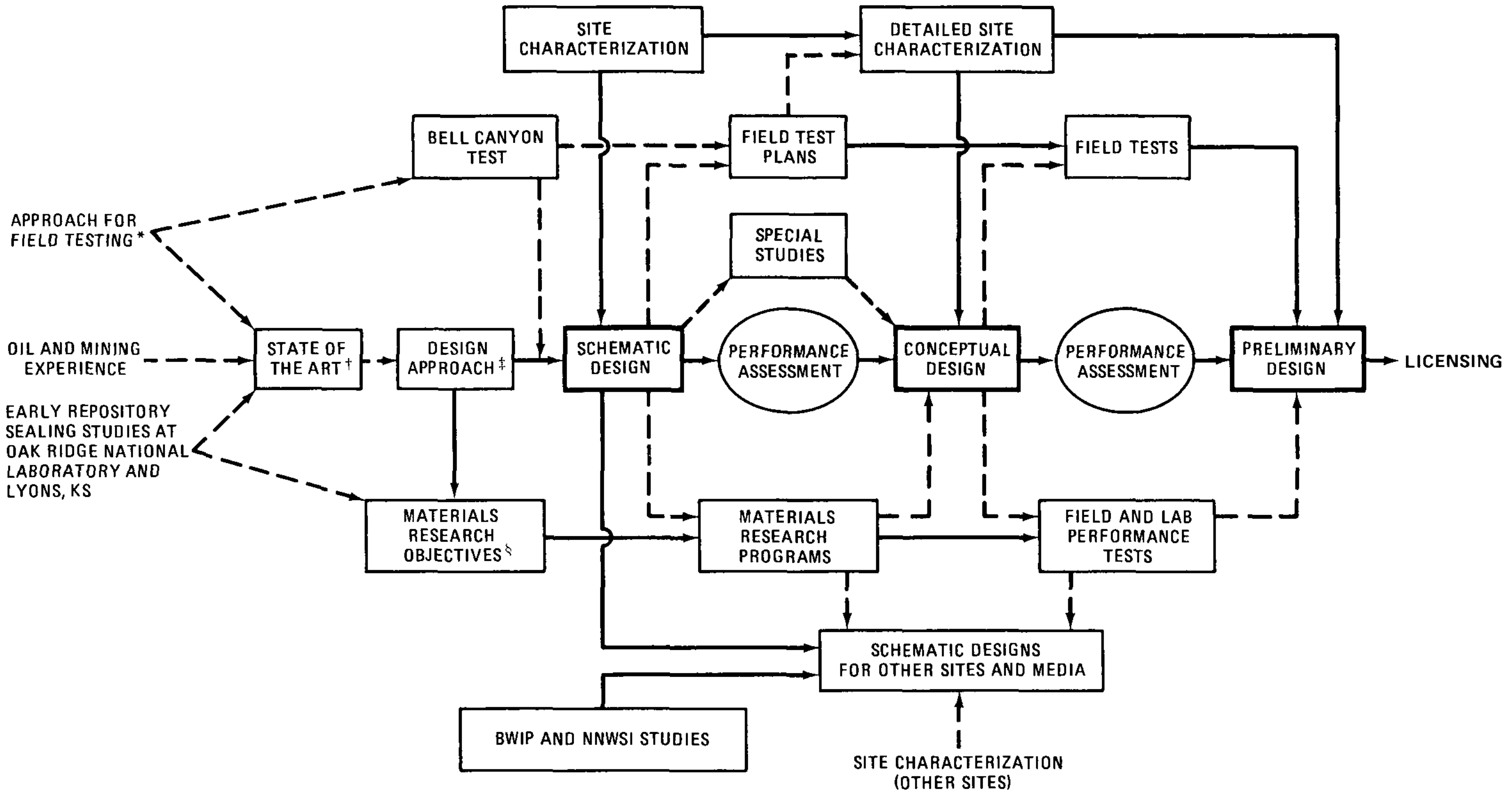

Figure 4 Design development flow chart.

**Office of Nuclear Waste Isolation (1978).*
†Office of Nuclear Waste Isolation (1979).
‡Office of Nuclear Waste Isolation (1980a).
§Office of Nuclear Waste Isolation (1980b).

ing parametric analyses of a wider range of conditions. Where possible, laboratory tests are preferred to field tests because they are less expensive and provide closer control of test conditions.

Parametric and sensitivity analyses are performed to evaluate seal performance and identify important elements. For example, an initial office activity was the determination of the impacts of the disturbed zone and the interface conditions on flow characteristics at a seal. These analyses permit efficient evaluation of variations of seal materials, interface and rock properties, and seal dimensions to determine the cost-effectiveness of various seal configurations. One of the preliminary results of the flow analyses is shown in Fig. 5. Similar analyses have been performed by Rockwell Hanford for penetration seals in basalt (Taylor et al., 1980). These types of analyses are now being expanded to also predict radionuclide retardation for a variety of materials, shapes, and conditions.

Current laboratory investigations include the plug tests conducted at

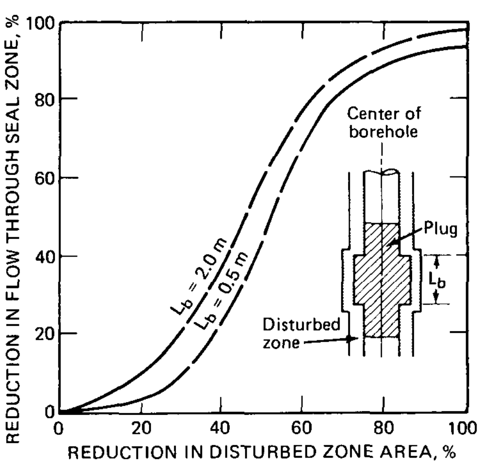

Figure 5 Influence of disturbed zone cutoff on flow through seal zone. Disturbed permeability = 100 × rock permeability.

Terra Tek, Inc., plus an extensive evaluation of candidate seal materials in progress at the Pennsylvania State University and the Waterways Experiment Station. Materials studies include determination of relevant thermomechanical, hydraulic, and chemical properties for candidate seal materials (Roy, Grutzeck, and Mather, 1980), studies of materials longevity based on consideration of thermodynamics (Sarkar, Barnes, and Roy, 1980), and examination of ancient materials (Langton and Roy, 1980). Completed studies within the ONWI program, including those referenced, emphasized cement-based materials. Future research will give further attention to other candidate materials, particularly clays. Materials studies for seals in the Hanford basalts examined bentonite-sand mixtures as possible seal materials. Other materials (such as polymer concretes, ceramics, and metals) will be evaluated to determine their potential as seal materials but will not be assigned priority for extensive testing.

The major field test conducted in the sealing program was the Bell Canyon test described earlier. At the present time, field tests and site characterization activities at candidate repository sites and test facilities are being monitored, but no specific repository sealing tests are in progress. Renewed emphasis will be assigned to field tests as in situ test facilities are developed at candidate repository sites. Field tests will be used to verify predicted seal performance and to evaluate seal emplacement techniques.

Schematic Designs for Penetration Seals at the Los Medanos Site

Site Geology and Hydrology. Detailed site characterization studies of the Los Medanos site have been conducted for WIPP (Sandia National Laboratories, 1978), with hydrologic data summarized recently by Mercer and Gonzales (1981). The proposed repository host rock unit is the Salado formation, a sequence of interbedded halite (approximately 90%), anhydrite, polyhalite, siltstone, and thin clay seams. At the center of the WIPP site, the Salado occurs at depths of 257 to 861 m. The proposed WIPP repository roof is at a depth of 655 m.

The Salado is overlain by siltstones and carbonates of the Rustler formation. The contact at the base of the Rustler is water bearing to the west of the site but is not believed to be water bearing within the site. For seal design purposes, however, the contact is regarded as a potential plane of weakness which might be a locus for future dissolution, forming a pathway to the biosphere. Two water-bearing dolomites, the Magenta and Culebra, occur within the Rustler. The Culebra is more permeable and occurs lower in the section. Permeability in the Culebra is controlled by fractures and vugs and can be highly variable. Groundwater in the Culebra and Magenta is unsaturated with respect to sodium chloride.

Stratigraphic and hydrologic parameters for the site are summarized in Fig. 6. The three water-bearing (or potentially water-bearing) zones in the Rustler represent key sealing locations in the respository shafts and boreholes both as potential pathways to the biosphere and as sources for unsaturated groundwater to enter the penetrations and dissolve salt at the seal interface.

Shaft and Tunnel Seal Components for a Repository in Salt. Seal designs are based on the principles that the shafts and the tunnels joining the shafts to the repository constitute a single seal system and that the seal system should include a number of different components, each of which is designed for a specific function (or combination of functions) and a specific location. Components are distinguished by a different geometry and/or a different composition, with each component matched to the environment in which it will perform. This concept minimizes many concerns arising from reliance on single materials that must perform for long periods under a variety of conditions. The multiple material-geometry concept provides: (1) high redundancy, which greatly reduces failure potential as a result of any uncertainties; (2) a variety of materials resistant to different environmental conditions; (3) means to improve interface and disturbed zones; and (4) opportunity to control both the quantity and chemical flow characteristics by sequencing selected geometries and materials.

It follows that the primary function of seal components will vary according to location within the system. In the shafts the primary function is to exclude water; this is achieved by placing effective seals in water-bearing zones and between water-bearing zones and the repository level. In the repository the primary function is to retard radionuclide movement; this is achieved by low permeability or by including highly sorptive or reactive seal materials. In the access tunnels joining the repository and the shafts, equal emphasis may be given to water exclusion and chemical retardation. This can be achieved either by incorporating seal components that have low permeability and high sorptivity or by incorporating separate components for each of the two functions.

Other seal functions must be considered also; for example, seal components are required to isolate panels within the repository and to provide structural support in the shafts and tunnels. Furthermore, consideration must be given to providing redun-

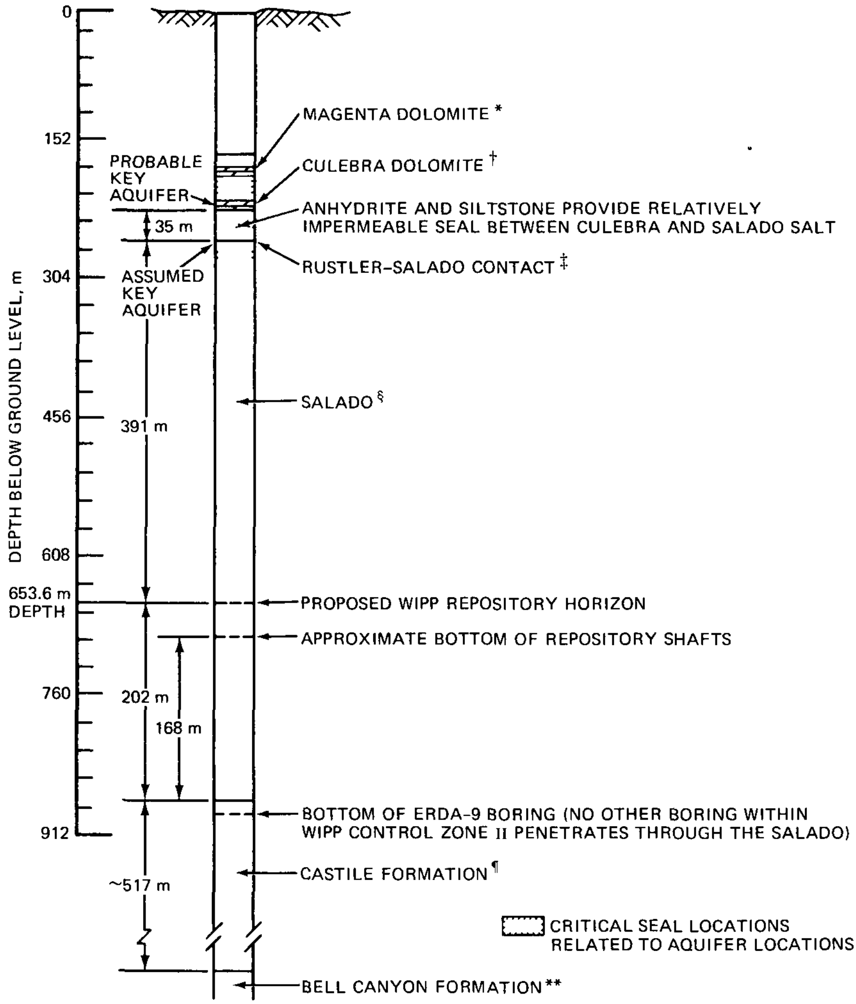

Figure 6 Shaft sealing considerations related to stratigraphy at the Los Medanos site. Stratigraphy to the 2886-ft depth is based on the ERDA-9 borehole. Permeabilities are predicted at shaft locations from existing borings within WIPP control zone II.

**Dolomite is more compact and less fractured than Culebra. Permeability is 10^{-7} to 10^{-5} cm/s.*

†This formation is vuggy dolomite, highly fractured. Permeability is locally variable (10^{-6} to 10^{-3} cm/s); this is the major aquifer in the system.

‡This formation is not water bearing at this site but is a plane of weakness which may control future dissolution. Permeability is 10^{-9} to 10^{-8} cm/s.

§This formation is halite (~90%) with anhydrite, polyhalite, sylvite, hydrous potash minerals, and clay seams. It is relatively impermeable but with permeability possibly greater along horizontal clay seams. Carnallite zones may be preferentially dissolved in the walls of boreholes or drilled shafts.

¶This formation is anhydrite with halite and limestone and is relatively impermeable.

***This formation includes sand aquifers with potentiometric surfaces above Rustler aquifer elevations.*

dancy in the system by incorporating components that have the same function but are made of different materials. Figure 7 shows a combination of seal components that might be considered for any repository in salt. Two basic types are included—backfills and bulkheads. Backfills are granular materials, compacted in place, and designed to have low permeability or high sorptivity or both. Bulkheads are structures keyed into the walls of the penetrations to intersect any disturbed zone that may result from excavation. Bulkheads will be constructed with stringent quality control and will be designed and emplaced specifically to achieve a low permeability interface with the host rock.

Materials Considerations. When choosing materials for seal components, we must take into account a number of factors, including the function of the component, the environment in which the seal will be placed, longevity requirements, suitability of the mined rock as a seal component, cost, ease of emplacement, ability to control quality, and the existing data base for candidate materials.

At any repository site it is desirable to use as much as possible of the mined rock in the seal system provided the crushed rock is suitable as a seal material. Crushed salt has a desirable property as a backfill material in a salt repository—it should tend to consolidate and anneal

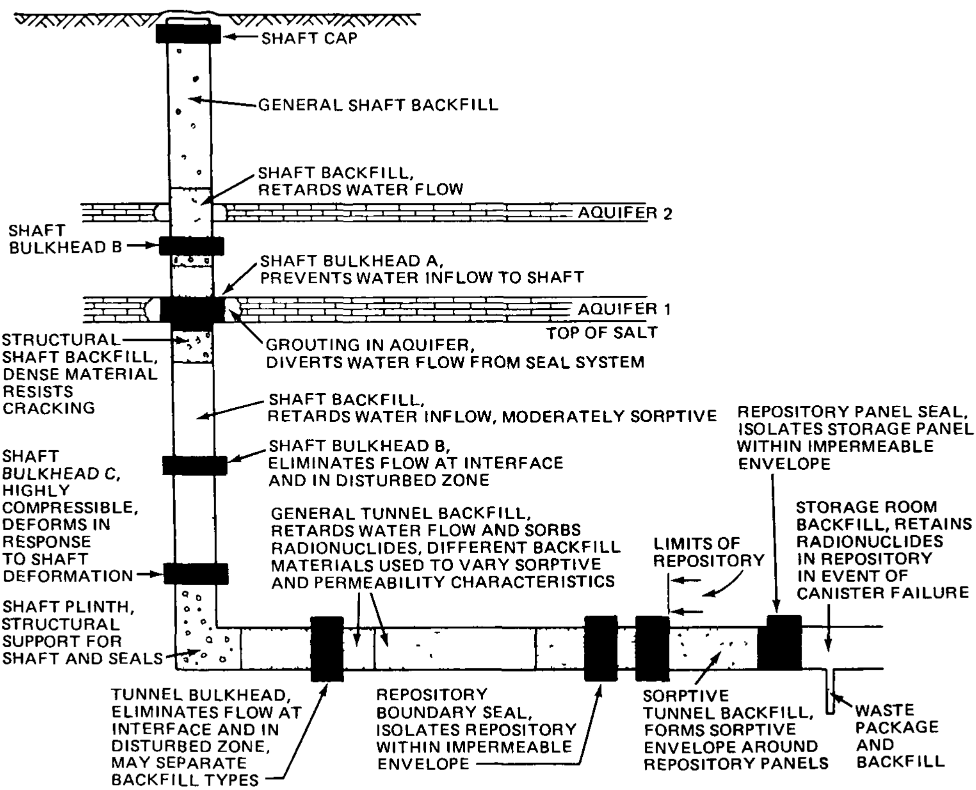

Figure 7 Essential seal components for shafts and tunnels for a repository in salt.

under the influence of pressure and elevated temperature, forming a homogeneous, relatively impermeable mass. Thus the waste might possibly be isolated within a salt monolith formed by fusion of the host rock and the salt backfill, which has a permeability similar to that of the undisturbed salt. This process may be hastened by using precast salt bricks in place of crushed salt. As a backfill material, however, salt also has a number of adverse properties; notably it is highly soluble and nonsorptive. Salt may be considered as a general backfill, except where contact with water is anticipated or possible (as in the shafts) and except where a high degree of sorption is required. The sorptivity of salt backfill may be increased, however, either by mixing the salt with a sorptive component such as clay or by interspersing sections of sorptive backfill within the salt.

A preliminary choice of suitable seal materials was made by comparing functional requirements with known properties for several classes of materials (Office of Nuclear Waste Isolation, 1980b). For practical reasons, priority has been given to two classes of materials for which a substantial data base is available—cements and earth (clay-bearing) materials.

Cements include cement mortars, grouts, concretes, and other portland-cement based materials, which are known to have a range of properties (permeability, durability, strength, etc.) consistent with the requirements for bulkheads and structural fills in the seal system. At present, two concrete types are being considered; one is dense and strong, and the other is sorptive and fracture resistant. The dense, strong concrete is similar in type and composition to the mix developed for the Bell Canyon test. It is composed of class H portland cement with silica fume and fly ash additives and a cement-to-aggregate ratio near 1:3. The concrete is slightly expansive, with unconfined compressive strengths near 138 MPa. Permeabilities are estimated to be near 1×10^{-10} cm/s (Gulick, Boa, and Buck, 1980). The sorptive, fracture-resistant concrete is composed of class H cement with a high proportion of bentonite additive. The resulting concrete may creep rather than fracture in response to stress or deformation. The bentonite component of the mixture also increases the sorptive capacity of the concrete and results in a low permeability.

Earth materials include crushed rock, gravel, sand, and clay, which can be mixed together as "earth fill" or graded to a narrow grain-size range. In general, natural materials that are compatible (possibly because they are mineralogically similar) with the host rocks at the site will be used. Clays are both sorptive and compressible, and appear suitable as major components for shaft, tunnel, and storage room backfill. Studies to evaluate the suitability of clays for repository sealing are in progress at D'Appolonia Consulting Engineers and the Pennsylvania State University.

Schematic Design for Los Medanos Tunnel and Shaft Seals. The schematic design for shaft and tunnel seals in bedded salt, referenced to the Los Medanos stratigraphy, is shown in Fig. 8. It is important to recall that this design will be reviewed and, if necessary, modified according to results from performance assessment and laboratory and field tests.

In the shafts the principal seal function is to prevent water from entering the seal zone. This is achieved using dense low-permeability fills, plus a number of bulkheads intended to eliminate flow at the interface and in any disturbed zone. Two types of fill are included—a gen-

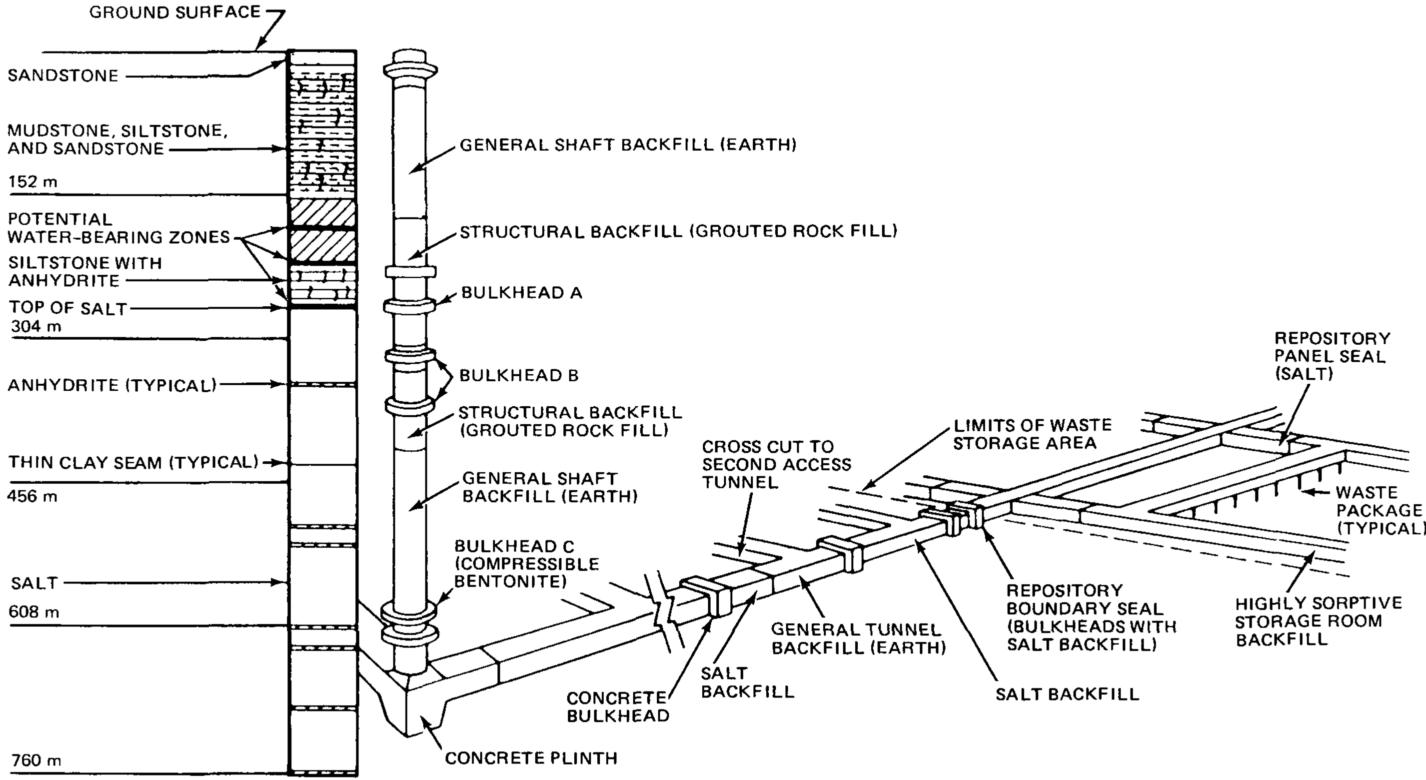

Figure 8 Schematic of seal system for shafts and tunnels at Los Medanos site.

eral backfill probably composed of earth materials (a clay-sand mix possibly with some cement) and a structural backfill composed of grouted rock fill. The latter will be placed with a high-density and low-moisture content. It is intended specifically to provide a low interface permeability and to be sufficiently strong to deflect stresses away from the bulkheads.

Three types of bulkhead are included in the shafts. Bulkhead A (Fig. 9) is placed at the lowest potential water-bearing zone, at the Rustler-Salado contact. A key element of Bulkhead A is the enlargement of the shaft at the contact zone and the construction of a specially formed concrete cylinder keyed into low permeability rock above and below the water-bearing zone. This cylinder will be backfilled with a well-compacted clay with very low permeability. Above and below the clay body are dense, impermeable fills intended to inhibit downward water migration through the seal zone. Depending on the rock conditions encountered in each shaft, the contact zone adjacent to the shaft may be grouted before the bulkhead is constructed. This area will be grouted regardless of any other grouting done during construction of the shaft unless the quality and longevity of the earlier work can be demonstrated.

Bulkhead B (Fig. 9) is a dense, impermeable structure keyed into the walls of the shaft, which is intended to prevent water flow through the disturbed zone and at the seal-host-rock interface. Bulkheads of this type will be placed between the Magenta and Culebra aquifers, immediately below the Culebra aquifer, at the bottom of the shaft, and at intervals in the shaft below the top of the salt as a backup to Bulkhead A. The bulkhead keys will be excavated cautiously, possibly by hand, and in small sections to avoid further disturbance of the host rock beyond the disturbed zone resulting from excavation of the shaft itself. The major portion of the bulkheads must be composed of a low-permeability material that bonds to the host rock, is stable, and leaves no connected void internally or at the seal-rock interface. A high-density concrete, which is slightly expansive, is planned for this material. Layers of compacted clay are located between the three concrete segments to avoid the potential for continuous crack formation through the concrete system as a result of temperature and/or deformation changes and to permit the bulkhead to be more flexible to deformation in the event of any outside disturbance.

Bulkhead C, located near the bottom of the shaft, is composed of highly compacted bentonite. This bulkhead will act as a backup water seal at the base of the shaft. Compacted bentonite is selected because of its extremely low permeability, because it will swell and tend to be self-sealing when wetted, and because it is compressible and will respond to any movement of the overlying seal materials or subsidence of the shaft walls. An added feature is that this component will be made entirely of "natural" materials.

The tunnel seals also include backfill and bulkhead components. Conceptually, the backfill will include interspersed sections of crushed salt and earth material. The crushed salt will be placed with a high density so that creep closure of the rooms will eventually form a homogeneous, impermeable mass. The earth material will be relatively impermeable but relatively sorptive also. The proportion of sorptive backfill will be increased in the disposal rooms, either by mixing a sorptive component with the crushed salt or by including more frequent sorptive sections interspersed with the salt.

Concrete bulkheads will be located at intervals in the access tunnels joining the shafts to the repository. One specific location for bulkheads will be at the entrance to the repository, where they will form "boundary seals." Each bulkhead will be composed of high-density concrete and will be keyed into the salt on all sides (Fig. 9). The length and shape of the bulkheads will be designed to control stress concentration effects, particu-

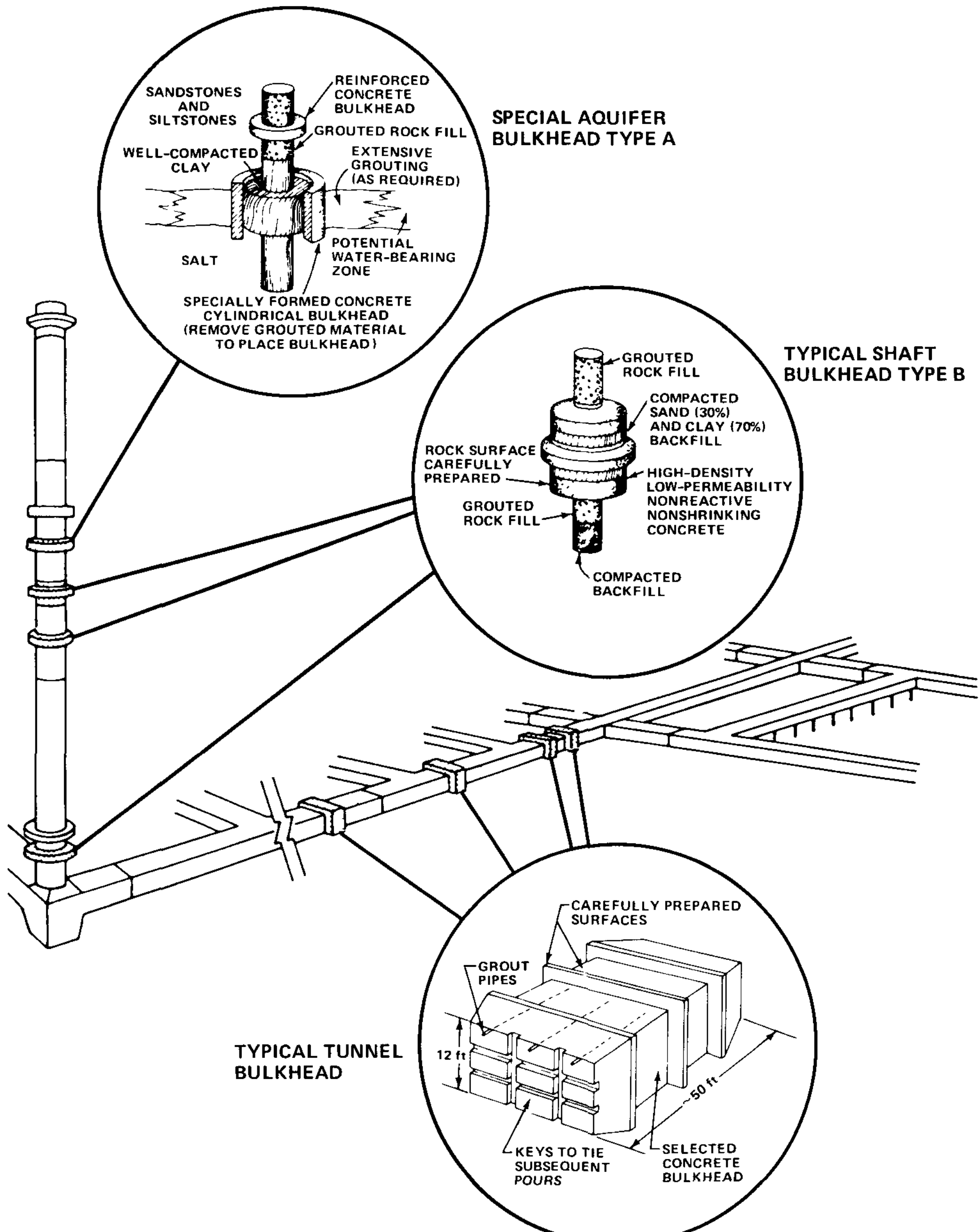

Figure 9 Bulkhead details.

larly near the center where the primary cutoff must occur. Grouting through the bulkheads will probably be necessary to improve interface conditions. In the long-term, interface permeability is expected to be further reduced by creep of the salt against the more rigid concrete.

Performance Assessment. An analytical assessment of the performance of the Los Medanos working design is in progress. Current assessments include: (1) total flow potential through the seal zone, (2) time of arrival for radionuclides, (3) potential radioactive release rates, and (4) impact of thermomechanical loading. Sensitivity analyses are also being performed for these studies. Future analyses will include: (1) geochemical modeling, (2) cost analysis, (3) time-dependent reliability (probabilistic) analysis, (4) creep closure vs. permeability, and (5) seal environment changes with time. An example illustrating the current analyses is presented in the following paragraphs.

Figure 10 shows results from a total flow analysis for the shaft seal system shown in Fig. 8. The model considers a steady-state flow, with the permeants injected at the bottom of the shaft seal and released to the biosphere at the top of the salt formation. The axisymmetric model represents a shaft with a radius of 1.8 m, a disturbed zone with a radius of 3.7 m, and a "no-flow" lateral boundary with a radius of 11 m. In this example, bulkheads are presumed to totally replace the disturbed zone.

The results for a 30.5-m driving head indicate a flow rate of 14 to 42 l/y for the base case with four bulkheads. This flow would be 1.5 to 4 times the flow through an undisturbed host rock with a radius of 11 m and the same driving head. Further analyses indicate that for high disturbed zone permeability (10^{-7} cm/s), removal of bulkheads can significantly increase the flow potential. Adding two more bulkheads does not significantly reduce the flow potential however. In other words, the presence of a disturbed zone has a major effect on system flow (as would be expected), but this effect is significantly reduced by including a small number of bulkheads that completely intersect the disturbed zone.

The average transit time of a water particle through the model in the example was estimated to be approximately 60,000 yr, assuming that the flow is occurring through a porous medium. This analysis is being extended to account for sorption, dispersion, and the presence of fractured media. Preliminary results indicate that fractured media could reduce the transit time by as much as an order of magnitude. On the other hand, even a weakly sorptive seal material would tend to significantly increase radionuclide transit time.

Future Activities

The conceptual design for bedded salt will be completed in fiscal year 1982. Schematic designs for other candidate repository media, notably dome salt, basalt, and tuff, have been initiated, and conceptual designs will be completed from fiscal years 1983 through 1985, as site characterization data become available. The basic design approach discussed here is anticipated to be applicable to all sites, although the detailed designs will vary greatly to suit site-specific conditions. An important activity early in this program will be to establish quantitative design bases and licensing requirements.

Materials testing will continue to focus on cement-based materials, but increasing attention is being given to clays. Scoping studies for other materials will be conducted to provide a broad data base for a variety of candidate materials. Increasing attention

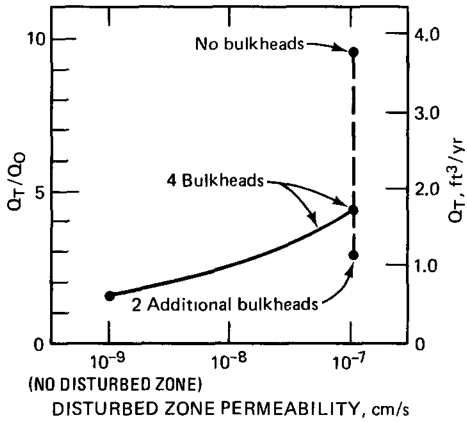

Figure 10 Flow through the shaft seal zone—Los Medanos Schematic Design: impervious boundary conditions (no flow across lateral boundaries). Q_T is total flow through the seal zone radius (11 m). Q_o is flow through the radius of undisturbed rock (11 m). Model height is 380 m; model boundary radius is 11 m; shaft seal radius is 1.8 m; and disturbed zone radius is 3.7 m. The hydraulic driving head is 30.5 m and is at steady state, saturated. Four bulkheads are included to cutoff disturbed zone. Material properties are:

	K, cm/s	Porosity, %
Host rock, salt	10^{-9}	0.8
Disturbed zone	10^{-8}	2.7
Anhydrite	10^{-8}	2.7
Concrete–gravel grout	10^{-9}	0.8
Earth fill	10^{-7}	35.0

will be given to materials compatible with host media other than salt. A wide range of properties will be evaluated, with particular attention to the characteristics of the seal-host-rock interface, seal-host-rock compatibility in anticipated repository environments, and longevity.

The materials testing program will be coordinated with an expanded program for testing full-scale borehole plugs in the laboratory. Additional tests, similar to those completed on a cement plug in anhydrite, will include cement and clay plugs in salt, basalt, and tuff. Laboratory tests to evaluate fracture healing in salt and consolidation and annealing of crushed salt will be performed also.

Field tests will gain additional emphasis in the sealing program when early shafts and test facilities become available at candidate sites in fiscal years 1984 and 1985. Tests may include a "bank" of borehole seals; this would enable a parametric evaluation of seal materials and emplacement methods and characterization of disturbed zones in shafts and tunnels. In the short-term, disturbed zone studies at generic sites (such as the test mine at the Colorado School of Mines) will continue and may be expanded. Field test plans for fiscal years 1982 and 1983 emphasize close coordination with site characterization programs to obtain basic data required for seal design.

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Borehole and Facility Sealing Activities for the Waste Isolation Pilot Plant

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The design of the Waste Isolation Pilot Plant (WIPP) proposed for a site in southeastern New Mexico includes a working level at 2150 ft with four shafts to the surface. About 70 holes have been drilled for site and mineral exploration in the 19,000-acre area being considered. Only eight of the holes penetrate below the repository level, however, and only four, all of them more than 1 mile from the underground workings, penetrate into the underlying aquifers. A development program is in progress at Sandia National Laboratories to provide adequate seals for these penetrations. Performance assessments indicated that effective permeabilities as high as 10 darcies do not result in doses to maximally exposed individuals greater than 0.01% of natural background. Materials have been developed, with emphasis on cementitious grouts, to match the WIPP lithologies. The grouts were evaluated in the laboratory, both alone and in contact with rock specimens, and in field tests. Results indicated that effective permeabilities of plugs measured in field tests (about 50 μdarcies), although still small, can be 100 times greater than the basic grout and 10 times greater than those observed in samples with the same rocks in the laboratory. Two major field tests, ERDA-10 and the Bell Canyon Test, were carried out, and a test series is planned which includes removal of an existing plug emplaced in 1976—a 26-in.-diameter hole was plugged, but with a central tube for diagnosing seal performance—and numerous tests in the experimental facility within the WIPP.

Introduction

Sandia National Laboratories is developing the technology for penetration sealing of nuclear waste repositories. This work, supported by the Office of Nuclear Waste Isolation (ONWI), addresses the sealing requirements of the Waste Isolation Pilot Plant (WIPP) and the proposed site in southeast New Mexico. The WIPP is specifically designed to demonstrate disposal of defense transuranic (TRU) wastes and as a research facility for experimentation on the interaction of defense high-level waste and bedded salt. The objectives of the penetration sealing program at WIPP are to (Christensen and Hunter, 1979):

- Assess the consequences of leakage through man-made penetrations on performance of the repository system
- Develop candidate materials for the WIPP stratigraphy
- Develop methods for assessing long-term geochemical stability
- Develop laboratory and field-testing techniques and to perform tests for evaluating emplacement techniques and plug performance
- Provide design information for plugs and emplacement techniques

Since activity is focused on site-specific applications in southeast New Mexico, materials and techniques that are compatible with the WIPP stratigraphy, hydrologic characteristics, and facility design are addressed primarily (Hunter, 1980). The technology developed in this program, nevertheless, supports the National Waste Terminal Storage (NWTS) program by providing materials, test techniques, and field data that can be applied explicitly to bedded-salt repositories and implicitly to all repositories.

Site Characterization. Geologic and hydrologic investigations have been in progress in southeastern New Mexico since 1974. The site presently under investigation, called the Los Medaños site and located in the Delaware Basin, is one of a series of sedimentary basins within the larger Permian Basin that extends into Texas, Oklahoma, Colorado, and Kansas. Extensive geologic and hydrologic investigations have resulted in a detailed characterization of the site under consideration (Powers et al., 1978).

The stratigraphy at the site is illustrated by the cross section shown in Fig. 1. The intervals of particular interest in the evaluation of the consequences of radionuclide release, and consequently the integrity of penetrations, are (1) Rustler Formation, at the center of the site, which extends from 168 m (550 ft) to 247 m (810 ft) below the surface; (2) the Salado, which extends from the Rustler to about 850 m (2800 ft) below the surface; (3) the Castile, extending from the Salado base to about 1220 m (4000 ft); and (4) the Delaware Mountain Group, which is about 1220 m (4000 ft) thick, with its top about 1220 m (4000 ft) below the surface.

The Rustler is primarily anhydrite, halite, and siltstone, but it does contain two dolomite beds—the Culebra and Magenta—which are the most sig-

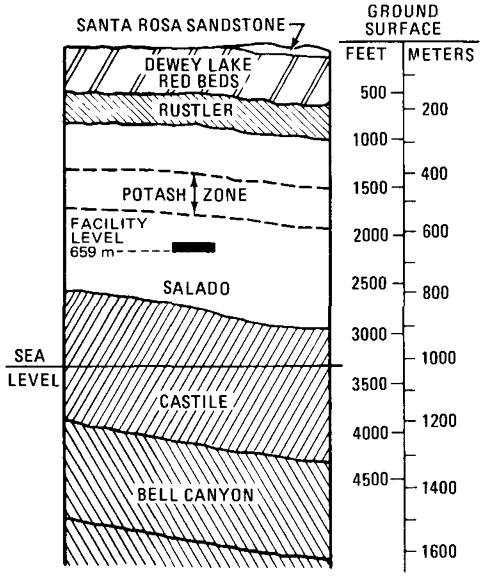

Figure 1 Stratigraphy at the WIPP site.

nificant aquifers in the area. Each of these beds is about 7.6 m (25 ft) thick at depths of about 220 and 186 m (720 and 610 ft), respectively. The Culebra and Magenta are confined aquifers with an average porosity of about 10 to 15% and a calculated transmissivity ranging from 10^{-10} to 1.5×10^4 m^2/s (10^{-4} to 140 ft^2/d), depending on the locality and degree of fracturing. The total dissolved solids range from 3,000 to 60,000 ppm (Powers et al., 1978). The proposed underground WIPP facility horizon (660 m, 2160 ft) will be located in the lower part of the Salado Formation. Near the disposal horizon, the Salado is primarily halite but has thin beds of anhydrite and polyhalite. Thin clay zones are also present. The Salado does not contain circulating groundwater.

The Castile is a thick (~400 m) anhydrite and salt formation underlying the Salado and forming an essentially impermeable barrier to the Delaware Mountain Group waters.

The Delaware Mountain Group, which consists primarily of sandstone, limestone, and shale, has been characterized as having a local porosity of 10 to 15% and a local conductivity of 7×10^{-8} m/s (0.02 ft/d). The upper formation in this group is the Bell Canyon Formation. Groundwater yields from wells in the Bell Canyon formation are approximately 3.8×10^{-5} to 9.5×10^{-5} m^3/s (0.6 to 1.5 gal/min) (Department of Energy, 1981) from isolated beds of sandstone about 20 ft thick.

Facility Design. The proposed facility for evaluating the disposal of defense wastes at the WIPP site is shown in Fig. 2. It consists of a single-level excavation with conventional room-and-pillar mining techniques. Surface facilities used during operation include a waste handling building, an underground personnel building for support of underground operations, a storage-exhaust-filtration building, an administration building, and various support buildings. The subsurface development includes four shafts to the underground, storage rooms for TRU wastes, and areas for experiments (Department of Energy, 1981).

Penetrations Associated with WIPP

Siting studies for a nuclear waste repository must consider the location of existing drill holes within the regions under evaluation. Sites for salt repositories are likely to be located in regions that have been subject to exploratory drilling for hydrocarbons or minerals such as potash ore. The identification of existing holes is essential in characterizing a repository site. In addition, site investigation will require the drilling of numerous holes to evaluate geologic and hydrologic characteristics or to further assess the potential for valuable minerals.

The WIPP site has been divided into four zones that will be controlled by the Department of Energy (DOE). Zone I consists of about 40 ha (100 acres) and will contain most of the surface facilities. Zone II, an area of about 730 ha (1800 acres), overlies the maximum extent of the potential underground development. Zone III, with a diameter of approximately 6.4 km (4 miles) and an area of 2500 ha (6200 acres), is the area in which drilling and mining will be precluded unless further evaluation indicates that it is acceptable. In zone IV, with a diameter of 10 km (6 miles) and an area of 4400 ha (11,000 acres), DOE may allow mechanical or drill-and-blast mining with certain restrictions, but not solution mining. In addition to compliance with requirements of the State of New Mexico, existing wells in zone IV may also require sealing by DOE-prescribed methods. New wells in zone IV would be drilled and sealed in conformance with these standards.

Boreholes. The most credible threat imposed by boreholes to the WIPP is development of communication between shallow and deep aquifers. The criteria used in identifying the current site included avoiding locations within 1 mile of any boring into the Bell Canyon or deeper formations (Powers et al., 1978). Consequently, the site does not include any deep holes within 1 mile of zones I or II.

Site characterization activities for WIPP in southeastern New Mexico included the drilling of 75 holes in the general vicinity of the site. Figure 3 shows the location of the exploratory drill holes and the existing industry holes that are within a 260-km^2 (100-mi^2) region surrounding the four zones that comprise the site.

The drill holes within the site, as a result of both existing industry and DOE exploration, can be generally

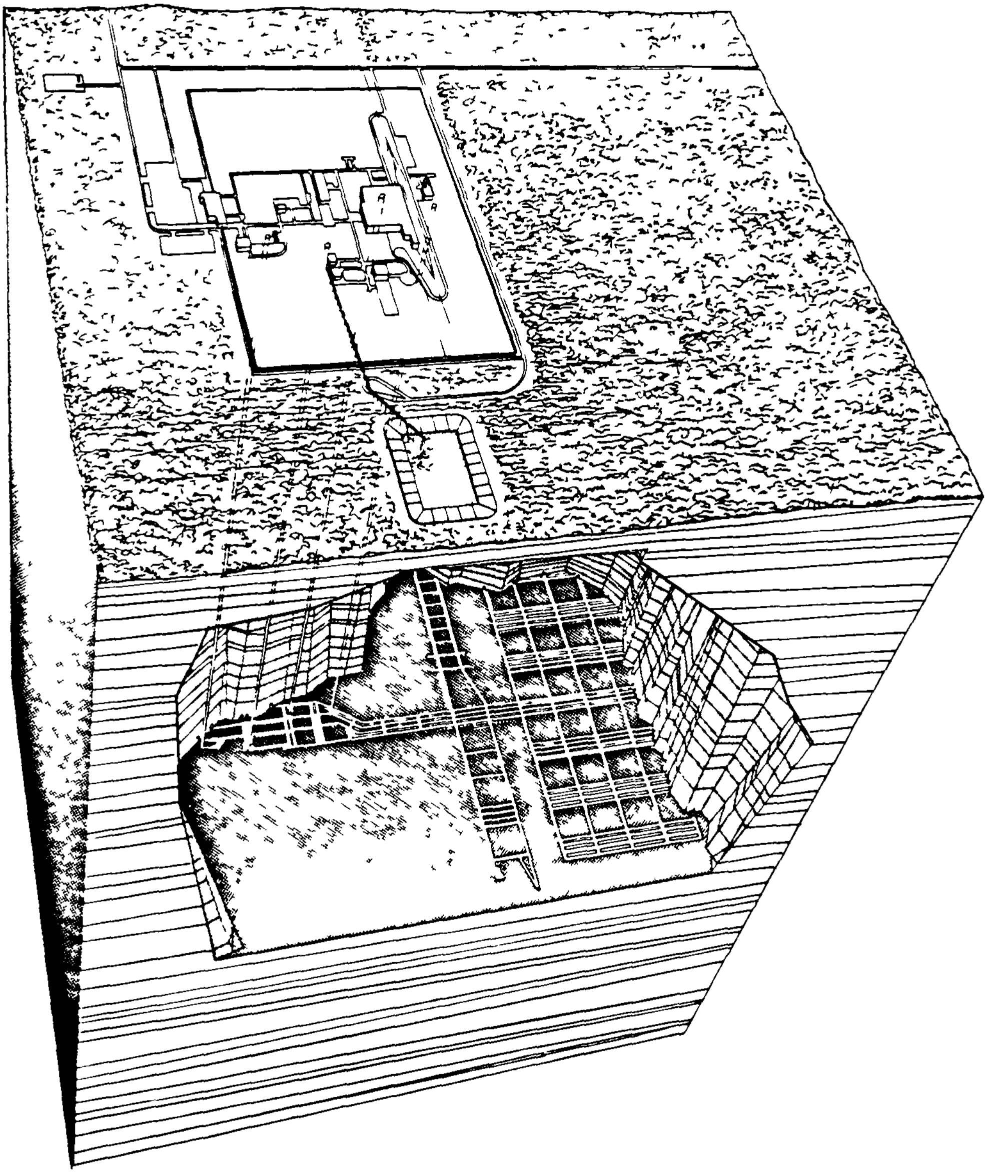

Figure 2 Proposed WIPP facility.

categorized as (1) geologic, (2) hydrologic, (3) potash exploration, or (4) hydrocarbon exploration. A total of 67 holes is present within zones I, II, III, and IV. Forty-six were drilled by DOE, and 21 were present when site investigation activities began. These drill holes are categorized by function and zone in Table 1.

None of the holes in zones I or II, shown in Table 1, extend to the Delaware Mountain Group (DMG). The deepest hole, ERDA-9, is 880 m (2886 ft) deep, which just penetrates the Castile Formation (861 m, 2825 ft) and stops 360 m (1200 ft) from the Castile/DMG contact. The holes listed in Table 1 are shown in Fig. 4

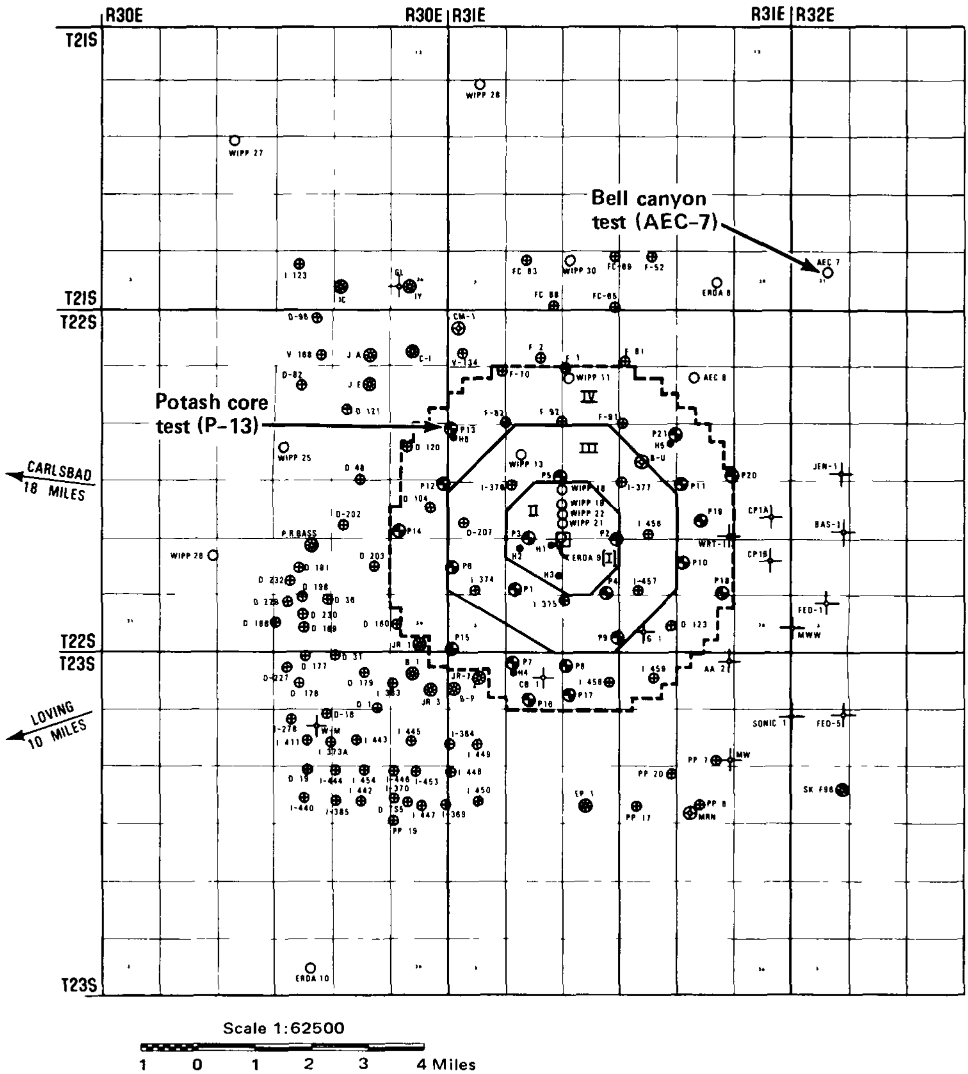

Figure 3 WIPP site drill hole location.

as a function of depth and distance from the center of the site. The only holes as deep as the disposal horizon in zone III are WIPP-12 and WIPP-13, geologic holes drilled by DOE near the northern perimeter of zone II and midway between the boundaries of zones II and III, respectively. An oil exploration hole (BWU-1) that penetrates 4600 m (15,000 ft), the deepest hole on the site, is located near the inner perimeter of zone IV approximately 1 mile from zone II. In all four zones, only eight holes penetrate as deep as the disposal horizon, and none of the remaining 59 extend closer than several hundred feet of that level. Thus the emphasis of the borehole plugging program at the WIPP site is concentrated on holes that extend to or through the disposal horizon since no mechanism is envisioned by which fluids can reach the disposal horizon from holes

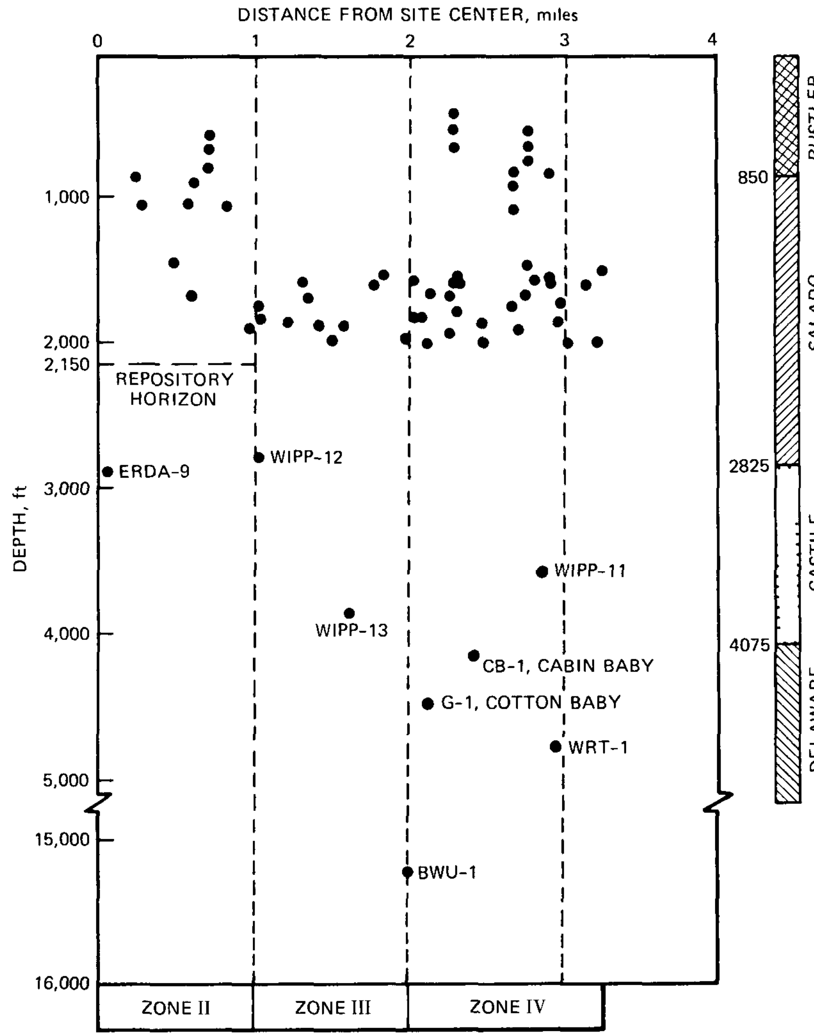

Figure 4 Depth of drill holes at WIPP site.

that end above it. Density stratification in a hole terminated in salt will not allow continued dissolution, and, moreover, the salt will ultimately seal the penetration.

Shafts. The design for the full WIPP facility includes four shafts, each extending to the disposal level (665 m, 2150 ft) and lined from the surface to the top of the salt (about 260 m, 850 ft), but unlined thereafter. These shafts are described as follows (Department of Energy, 1981):

1. Waste Shaft, 5.8 m (19 ft) in diameter, for transporting waste
2. Ventilation Supply and Service Shaft, 4.9 m (16 ft) in diameter, for transporting personnel, materials, and equipment and for primary air intake
3. Construction Exhaust and Salt Handling Shaft, 4.3 m (14 ft) in diameter, for salt removal and exhaust air duct for underground construction areas
4. Storage Exhaust Shaft, 3 m (10 ft) in diameter, tne exhaust air

Table 1 Categorization of Drill Holes Within the WIPP Site

Zone	Geologic	Hydrologic	Potash	Hydrocarbon
I	2 (DOE)			
II	4 (DOE)	5 (DOE)	2 (DOE)	
III	2 (DOE)		5 (DOE) 7 (Industrial)	
IV	3 (DOE)	9 (DOE)	14 (DOE) 10 (Industrial)	4 (Industrial)

duct for underground waste storage areas

Site validation activities at the WIPP include plans for early development of two shafts for underground investigations and in situ experimentation. These two shafts will ultimately be converted into the storage exhaust shaft and the waste shaft when construction of the facility is completed. The waste shaft would be bored initially at 1.8 m (6 ft) in diameter and subsequently expanded to full size. All shafts will be completely plugged when operations cease. One of the shafts would also be coincident with drill hole ERDA-9, which currently penetrates through the Salado formation, thus eliminating concern over two penetrations at that location.

Program Elements

The penetration sealing program is composed of four integrated elements to provide an adequate and demonstrated technical basis for design specifications: performance assessment, material development, instrumentation development, and field tests.

Performance Assessment. The significance of seal integrity has been evaluated in consequence assessment scenarios performed for the WIPP, which include assumptions about breaching of the media surrounding the repository and estimations of the potential dose to human populations.

A series of these scenarios were identified for WIPP (Bingham and Barr, 1979), and consequences of the radionuclide release were determined and published (Department of Energy, 1980, 1981). Three of these scenarios are referred to as liquid breach and transport scenarios and can be examined to determine the role played by borehole plugging:

1. An open (i.e., unsealed) borehole that penetrates the repository allowing fluid transmission between lower and upper aquifers
2. Two interconnected penetrations of overlying formations that allow fluids to enter the repository at one location and escape into the upper aquifer at another
3. Diffusion of radionuclides from a flooded repository into overlying aquifers

In all analyses for WIPP, no credit is given to the stability of the waste form beyond an assumed dissolution rate equivalent to that of the salt in which the waste is located. Radionuclide transport in the aquifers is based on retardation parameters (sorption), which are derived from laboratory data, and no credit is taken for sorption in the salt or associated minerals (Dosch and Lynch, 1978). Fluid movement is based on prevailing regional groundwater hydrology determined by field tests at selected points. Transmissivity of the aquifers was varied, and consequences were based on maximum values measured for a

Table 2 Results of Consequence Assessments for a TRU Repository

	Penetration assumption	Highest annual total body dose rate, mrem/yr	Greatest organ (bone) dose rate, mrem/yr
Scenario 1	Open 9-in. borehole (10^5 darcy)	7.7×10^{-3}	1.3×10^{-2}
Scenario 2	Two 24-in. boreholes (20 darcy)	1.7×10^{-3}	2.8×10^{-3}
Scenario 3	50-acre open cross section to aquifer	7.0×10^{-5}	1.2×10^{-4}

given region. Doses are calculated for the maximally exposed individual.

Results from the consequence analyses for the maximum scenarios for WIPP, which contained only TRU wastes (Department of Energy, 1980, 1981), are presented in Table 2, along with the associated assumptions about penetration integrity for each scenario.

These results indicate that scenario 1 yielded the highest dose to individuals (7.7×10^{-3} mrem/yr, whole body), which is about 0.005% of the dose from natural radiation in the United States.

Conclusions from these studies indicated that borehole seals with effective permeabilities on the order of tens of darcies would result in doses to maximally exposed individuals of less than 0.01% of natural background. A conservative approach and the use of multiple isolation barriers, however, require that penetrations be sealed to the extent practicable to preclude the events that allow fluid intrusion and potential release of radionuclides, albeit of small consequence. Further, it is important to assess quantitatively the ability of seals or plugs to restrict fluid flow so that a measure of the additional protection they provide can be determined.

In addition to overall assessments, techniques have also been developed to evaluate the effectiveness of a plug-formation system to block fluid flow along the axis of the borehole. These techniques, which allow us to characterize the flow region within or surrounding a plug, have been applied to results from tests in which flow was measured around an in situ plug (Peterson and Christensen, 1980).

Materials Development. Materials are being evaluated to measure their adequacy for repository sealing and efficient emplacement. Current emphasis has been on evaluation of cementitious grouts. Tests have been in progress for several years to determine what materials are compatible with rocks at the WIPP site and exhibit the characteristics necessary for a plug (Gulick et al., 1980). Grout mixes with expansive cement have been developed with various additives and both brines and fresh mixing water.

Some desirable properties for plugging grouts are low permeability, low porosity, high density, high strength, expansive potential, isotropy, homogeneity, pumpability, adequate working time, stability, and durability. Many of these properties can be achieved by reducing the water-cement ratio in grouts made with portland cement products. Pumpable grouts generally require far more water than is necessary for complete hydration of the cement. Reduction in water-cement ratios has been studied and was achieved by varying the coarseness of the grind of the cement; using water reducers (including high-range water reducers called superplasticizers), turbulence-inducing compounds, and/or retarders; and varying the temperature of the grout slurry. Lowering the water-cement ratio generally reduces permeability and porosity and increases density and strength (Gulick et al., 1980).

Expansive cement systems were included to provide a positive expanding force against the rock surface of the borehole after it hardens. This force should reduce the number of microfractures in the stress-relieved annulus of rock around a plug and provide for a tighter interface contact between the plug and the rock. Improved bonding should resist plug movement and decrease permeability of the plug, particularly at the interface.

Both fly ash and natural pozzolans were included in the studies to determine their effects on porosity, permeability, and durability. Fly ash was included in all mixtures (30% by solid volume replacement for cement) to reduce early age temperature rise and to contribute to later age strength and durability by combining with calcium hydroxide to form the more stable and durable calcium silicate hydrates (Gulick et al., 1980).

Salt (sodium chloride) was included in some grout mixtures because of the evaporite rock sections at the WIPP site. Brine prevents the salt rock at the interface from dissolving while free water is present in the grout.

Cements low in tricalcium aluminate (C_3A), which are generally used in the Southwest, were the basis of the grout mixtures. Low C_3A content is a major factor in resisting sulfate attack (Gulick et al., 1980). Fly ash is also a factor in improving resistance to sulfate attack. Using the lowest possible water-cement ratio, because of its effect on gel-pore structure of grout, can also significantly improve stability and durability of grout plugs (Gulick et al., 1980).

Grout formulations were prepared for field testing applications, in which they were designed to be compatible with a specific location in the WIPP stratigraphy. The most recent example of this development is that for the Bell Canyon Test (Gulick, Boa, and Buck, 1980); it resulted in two baseline grout formulations for WIPP-BCT-1F, a saltwater-based mix for rock salt sections, and BCT-1FF, a freshwater mix used for nonsalt sections.

The characteristics (e.g., permeability and bond strength) of these mixes and associated rocks were determined in the laboratory and then compared with in situ performance. Laboratory tests included evaluating the grout alone and then in combination with the host rock. A specimen of grout cast in an anhydrite core from the site of the Bell Canyon Test is shown in Figs. 5 and 6. The performance of grout or a sealing material can be measured qualitatively by comparing the effective permeability of cured grouts under the three different test conditions. These data (Gulick, Boa, and Buck, 1980; Christensen and Peterson, 1980) are presented in Table 3.

We can see that the important influences are not only the basic matrix permeability of the grout but also the interaction with the host rock and emplacement environment.

Studies of grout feasibility have been coupled with strategies for assessing long-term stability (Lambert, 1980c). Samples of plugs in potash-exploration holes were also taken from mines located near the WIPP site. Observations revealed that no serious degradation had occurred after 17 yr of curing at a depth of

Figure 5 Simulated borehole sample to determine grout performance.

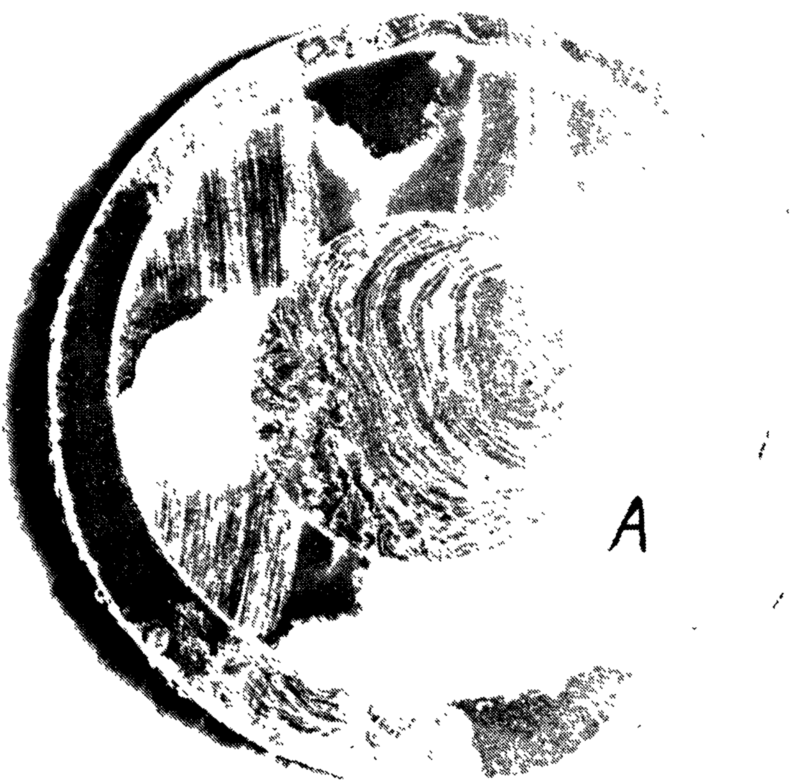

Figure 6 Cross section of simulated borehole sample showing grout and anhydrite.

Table 3 Effective Permeability of BCT-1FF (10^{-6} darcy)

Bare grout	Cast in anhydride	In situ
<0.1	~1-5	~50

approximately 1000 ft and that permeabilities were in the millidarcy range (Buck and Burkes, 1979). The x-ray diffraction patterns and scanning electron microscope analysis of the 17-yr-old seal showed similarities in composition and microstructure to those of the ERDA-10 grout at 2 weeks and 1 yr. There was evidence of relatively little exchange or reaction between the cement plug and the surrounding rock over the 17-yr period (Scheetz et al., 1979). A cooperative program with Sandia, ONWI, Pennsylvania State University, and the Army Corps of Engineers Waterways Experiment Station has been formulated to further assess the potential for the long-term integrity of plugging materials.

Natural materials have also been considered as candidates for repository sealing. These materials have the apparent potential for long-term stability because of their low reactivity with the host rock. Principal candidates include salt (Butcher, 1980), either crushed or in large blocks, and clays (Nowak, 1980). A solution-precipitation technique has also been identified as a possible method for emplacing salt in vertical penetrations within the salt (Lambert, 1980b). A combination of tailored grout and natural materials is expected to provide adequate sealing for the facility. Grouts should provide competent seals until reconsolidation and compaction of natural seals occur and may even have the potential for long-term sealing, depending on their longevity.

Instrumentation Development. Instrumentation has been developed to evaluate the environment in which plugs are placed and to determine the performance of an emplaced plug (Cook, 1979). Developments to support a specific test include (Cook et al., 1980):

- A guarded straddle packer system to evaluate formation permeability
- A self-contained instrumentation package incorporating a timed-release tracer gas, temperature and pressure measurements, and packer-release system. When this package was emplaced below a plug, it functioned without any communication from the surface
- A geophone system to monitor well-bore acoustic sources and instrumentation functions
- Fluid buildup and shut-in pressure measurements using current oil-field systems
- Probes for measuring discrete fluid leaks, fluid electrical conductivity, pressure, and temperature
- A wire-line closed-circuit TV system capable of operating in drill holes to a depth of 1500 m (Christensen, Statler, and Peterson, 1980)

Field Tests. Field tests to evaluate emplacement techniques and the performance of plugging materials under in situ conditions are continuing near the WIPP site. Data on the performance of plugging techniques from the completed tests are correlated with the laboratory data discussed and ultimately are used to assess the performance of similar techniques in a repository configuration.

In 1977 an exploratory drill hole, ERDA-10, was plugged to develop field emplacement techniques for grouts and to initiate a quality assurance program (Gulick, 1979). Three grout mixes were formulated for specific locations, and four separate plugs were emplaced for a total length of 1350.26 m (4430 ft).

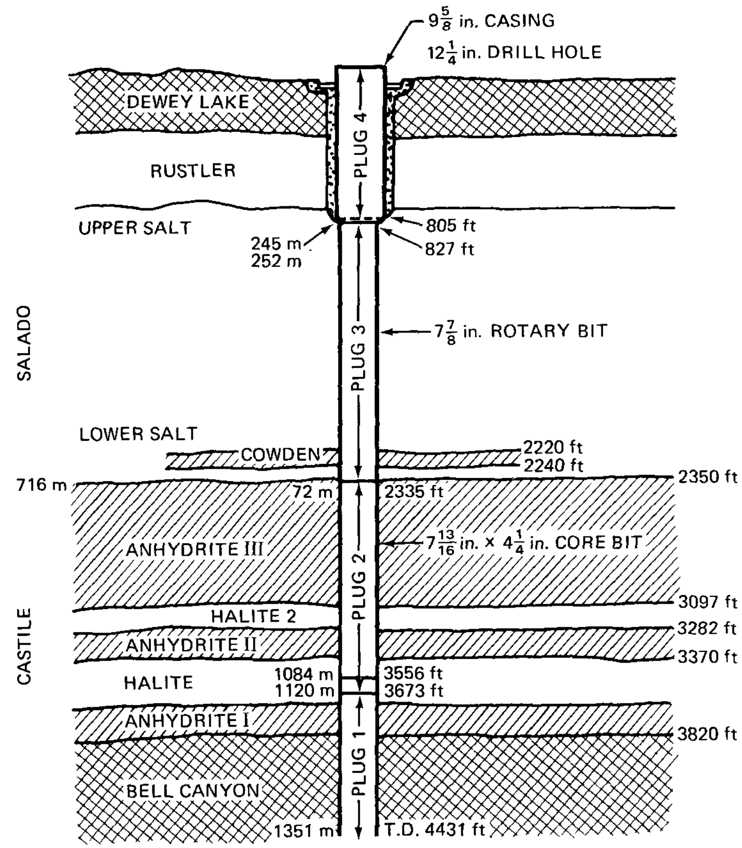

Figure 7 Plugging configuration of ERDA-10.

Samples of grouts cast during emplacement operations and cored from these plugs after they set up were obtained for laboratory analysis. The plugged configuration for ERDA-10 is shown in Fig. 7.

In a more recent field test, the Bell Canyon Test (Christensen and Peterson, 1980; Christensen, 1979), which was completed in a drill hole near the proposed WIPP site, a grout plug 2-m long was placed in a thick anhydrite bed at a depth of 1370 m. The plug was located directly above an aquifer that provided a 12.9-mPa (1800-psi) pressure differential across the plug. The aquifer had a production capacity of 38,000 l/d. After the plug cured, a series of tests was performed to evaluate the impedance to flow it provided. The observed leakage was 0.6 l/d, which was equivalent to a plug with an effective permeability of 50 μdarcies. Laboratory results and analysis of field data lead to the conclusion that the bulk of the flow occurred at the plug-host rock interface (Christensen and Peterson, 1980). After testing of the initial plug was completed in February 1980, an additional 4 m of grout was emplaced immediately above the first. Testing of this plug is expected to be completed in the near future.

Another test, the potash core test (Christensen, 1980), is in the final stages of preparation. In this test a plug (P-13) that was emplaced in 1976 will be selectively overcored as shown in Fig. 8. The existing plug (8-in. in

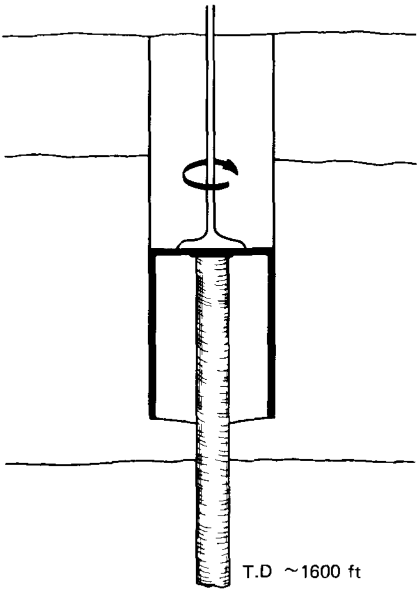

Figure 8 Schematic of the potash core test.

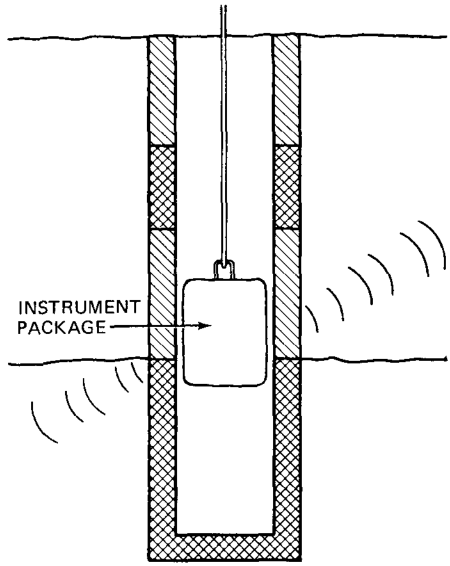

Figure 9 Schematic of the diagnostic test hole.

diameter) will be removed within a 14-in.-diameter core, which will subsequently be taken to various laboratories for evaluation of compatibility between the plug and the host rock and the effectiveness of the seal (Lambert, 1980a). Test and operation plans, site preparations, and core handling equipment for this test have been completed. The core barrel, which is 20 in. in diameter, is one of the largest ever fabricated. Field operations have been delayed, however, pending resolution of program uncertainties.

After the potash core test is completed, another test, the diagnostic test hole, is planned for the same hole. The 20-in.-diameter hole will be reamed to 26 in., and a plug will be installed with a central tubing, as shown in Fig. 9. The central tubing will allow access for instrumentation to monitor grout properties and fluid movement at various locations.

Additional tests for repository sealing will be performed at potash mines near the WIPP site. In situ tests will also be performed at the repository horizon for WIPP as part of the site and preliminary design validation program, in which two shafts will be sunk to 2150 ft and underground rooms will be provided for experiments. The shaft drilling began in July 1981, and access to underground areas for sealing experiments is expected in 1983.

Summary and Conclusions

The development of the Waste Isolation Pilot Plant in southeastern New Mexico includes a program addressing repository sealing, which will ultimately provide the design, materials, and technology for sealing shafts, boreholes, and repository rooms. Development work included performance assessments to provide a basis for the desired sealing integrity. The assessments indicated that consequences to human populations in the future are not significant even in the case of completely open penetrations.

Nevertheless, seals with the highest integrity practicable will be provided to further increase the confidence in waste isolation.

Materials, especially cementitious grouts, have been developed which are compatible with the WIPP lithology. These grouts are under evaluation in both laboratory and field tests. Natural materials have also been identified as candidates for long-term stability. Field tests have been performed, and others are planned in the near future. Supporting instrumentation has also been developed. All information available to date indicates that adequate seals can be provided by using a combination of materials and that their adequacy can be evaluated by laboratory and field tests.

Acknowledgment

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Design Approaches for Access Plugs in a Basalt Repository

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This paper describes research, laboratory testing, and analytical approaches taken toward the development of designs for sealing boreholes, shafts, and tunnels penetrating from ground surface to a deep, mined nuclear waste repository in basalt. A material selection process leading to identification of preferred sealing materials is discussed, and the laboratory testing program to define the geochemical and geotechnical performance of these materials is described. Analysis of the environmental conditions in the Columbia Plateau basalt flows leads to identification of tentative design criteria for plug systems. These design criteria include performance of the plug as a hydraulic barrier and as a radionuclide barrier. An important problem for effective performance of a plug system as a hydraulic barrier is shown to be a potentially disturbed zone surrounding the excavation in the stressed and jointed host rock. An idealized one-dimensional numerical model is described for analyzing the performance of the plug as a barrier to radionuclide transport. The preliminary analyses led to the conclusion that the composition and dimensions of practical candidate plugs can satisfy both hydraulic and radionuclide barrier criteria. Examples of candidate designs are shown for boreholes, shafts, and tunnels. 9 ref 6 fig 6 tab

Introduction

The preliminary work for the preconceptual designs of plugging systems described in this paper was begun by Woodward-Clyde Consultants in 1979 in support of Rockwell Hanford Operations (RHO) Basalt Waste Isolation Project (BWIP), being conducted by RHO for the Department of Energy. In the first phase of work, the geologic, hydrologic, and geochemical features of the Columbia Plateau basalt flows relevant to borehole plugging performance were characterized. Then, by a screening technique, approximately 10 candidate plug materials judged suitable to the repository environment were selected for laboratory testing and preconceptual design studies. Preliminary screening criteria included consideration of chemical stability in the repository environment, documented examples of stability over the period of performance required (assumed to be 10,000 yr), and performance attributes, including ability to inhibit fluid flow, inhibit radionuclide migration, and provide structural integrity in a plug. (See Fig. 1 for an example of the material selection process.) The criteria favor unprocessed, natural material because, in general, they can be proved to have existed in the same state for geologic times in stable condition and in environments the same or similar to the basalt environment. Nevertheless, some processed natural material (e.g., cement) was admitted,

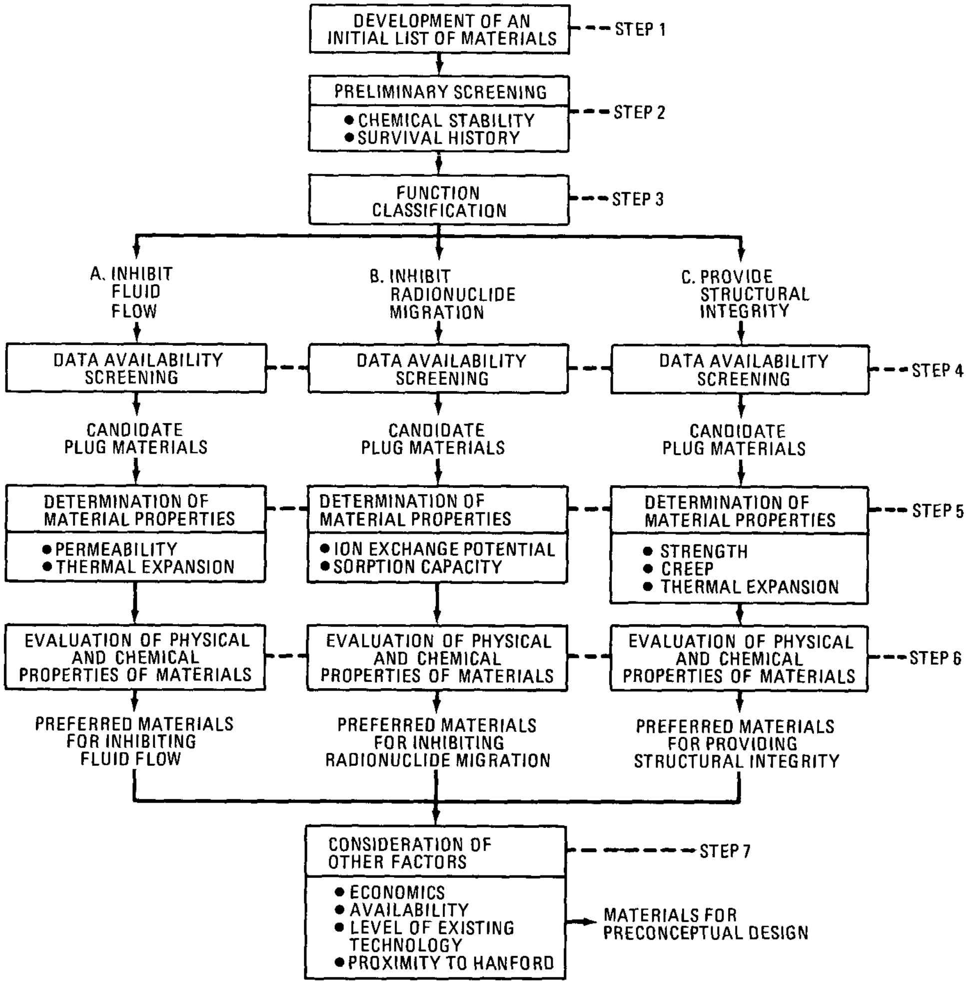

Figure 1 Material selection process.

despite the difficulty of documenting its stability in time, because it appears to meet environmental needs and has excellent performance characteristics. The selected materials included coarse- to fine-grain basalt; Oregon and Wyoming bentonites; processed bentonites; Ringold clay; Oregon zeolite; glaciofluvial sand; and type II, type V, calcioaluminate, and type K portland cements.

Testing

Immediately after the preferred candidate plug materials were identified by the screening process, geochemical and geotechnical testing of the material properties was begun. Tests of the geochemical properties at temperatures to 250°C and pressures to 34.5 MPa indicate that the candidate plug materials are compatible with each other and with the geochemical environment expected near a repository in Columbia River basalts. The physical testing program was designed to aid in selecting individual materials and mixtures from the list of preferred candidate materials and to provide descriptions of the physical properties of these materials for use in design studies. Preliminary results indicated that, of the cements, type V portland cement is preferred for use in plugs because of its high sulfate resistance and good mechanical performance. Adding finely ground, high-silica pozzolan to portland cement reduced shrinkage, increased workability, improved impermeability, and increased stability of cement mixtures exposed to moderate temperatures (less than 100°C). Adding finely ground silica flour reduced shrinkage and substantially improved structural strengths of cement mixtures exposed to temperatures greater than 100°C.

Preliminary results for cohesive soil materials indicated that bentonite clay is a preferred material for use in clay-sand or skip-graded clay-sand-aggregate plug mixtures. Its high plasticity, excellent swelling properties, and very low permeability, coupled with high ion-exchange potential and sorption capacity, should substantially decrease both fluid and radionuclide migration through plugs. Wyoming bentonite is superior to Oregon bentonite in terms of plasticity and impermeability, and both are far superior to Ringold clays from the Hanford site. Both crushed basalt and glaciofluvial sand and gravel from the Hanford site are strong, competent granular materials suitable for use as aggregate in concrete, compacted earth materials, and premixed clay slurries. At the present time, crushed zeolite appears useful only as a component in compacted backfill.

Preliminary tests with miniature plug models (plug materials are emplaced in holes drilled into blocks of basalt) indicate that it is possible to design mixtures of candidate plug materials having a permeability of less than 10^{-8} cm/s and forming good bond strengths with the host rock. Mud contamination of simulated borehole walls during testing was found to substantially decrease the bond strength between miniature cement and soil plugs and the basalt. Interestingly, high bond strengths for compacted bentonite-sand mixtures cured at 100°C indicate the possibility of cementation between plug and basalt at high temperatures.

Preconceptual Design Criteria

Considering the principal objectives in the performance of a plug led to the recognition that it should be designed and analyzed as both a seepage barrier and a radionuclide migration barrier. Theoretically, the plug might be designed strictly as one type of barrier or the other to achieve a competent, waste isolation plug system. For example, if a plug could be designed to allow absolutely no fluid to seep through it, there would be no need to consider its potential as a radionuclide migration barrier, assuming fluid transport is the only mechanism of mobility available to the radionuclides. On the other hand, if the plug was designed to have properties that would trap the radionuclide content of the seepage flow no matter how much seepage took place, there would be minimal need to consider its seepage barrier properties. In actual practice, however, both the seepage barrier and the radionuclide barrier properties of the in situ plug will be difficult to quantify exactly; thus it is safer to design the plug to meet both types of barrier performance criteria.

Essentially, two types of seepage criteria were considered initially:

1. Allowable seepage through the plug, plug-basalt interface, and excavation-disturbed wall rock. This approach specifies a maximum annual seepage rate. Note that any instability of the plug which causes cracking or failure will increase seepage. Thus the criterion implies a condition of plug

stability. The allowable seepage was considered to be 1 m^3/yr for 10,000 yr.

2. Seepage travel time through the plug, plug-basalt interface, and damaged wall rock. This approach specifies how long it must take for one drop of water to traverse the plug. A travel time of 10,000 yr was considered. This is a more restrictive criterion than allowable seepage.

It was decided to adopt the criterion of 1 m^3/yr seepage and to proceed with the preliminary feasibility analyses for the plug, considering that it might be constructed of any one or more of the 10 candidate materials, and then check the feasibility of the plug's performance as a radionuclide migration barrier.

Seepage Analysis

The quantity of water flowing through the plug system was computed using Darcy's law for flow through a porous medium, which states:

$$Q = kiA \tag{1}$$

where Q = quantity of flow
k = coefficient of hydraulic conductivity
i = hydraulic gradient
A = cross-sectional area

For this analysis, flow is flow through the plug and the disturbed basalt around the plug only; any flow through the undisturbed basalt was not considered. This is a valid assumption if the permeability of the plug and/or the disturbed rock is significantly greater than that of the undisturbed rock.

The worst-case condition for flow moving out of the repository is assumed to be a pressure head of 160 m of water, which is approximately the saturation pressure of the backfill pore fluid changed to steam at 200°C. This condition may exist for a time after the repository is sealed, until the groundwater system recharges above 160 m of head. The flow is assumed to be steady state; this means that the head has remained constant over a long enough period of time for the flow to stabilize, and the plug is saturated. The permeability of undisturbed basalt is estimated to range from 10^{-13} to 10^{-3} cm/s, with an assumed value of 10^{-9} cm/s for the basalt formation in which the repository will be constructed. Work by Iwai (1976) on the permeability of fractured basalt showed minimal changes in permeability for confining pressure changes above 5 MPa, but an increase in permeability by two orders of magnitude occurred when confining pressures were decreased from 5 MPa to 0. This would result in a permeability of 10^{-7} cm/s for fractured basalt with low confining pressure. If the permeability is greater than 10^{-7} cm/s, grouting may be able to reduce it to that value. The thickness of the zone of fractured basalt around a plug is expected to depend greatly on the construction techniques used in excavating and supporting the tunnel.

On the basis of in situ stress measurements made by overcoring techniques in underground excavations and the measured depths through the excavation-disturbed stress zone, rock depths ranging from 0 to 2 times the tunnel diameter were investigated in a parametric study, assuming a 10^{-7} cm/s permeability for the disturbed rock area.

Two values of permeability were considered for the plug material—10^{-8} and 10^{-9} cm/s, both of which fall within the range of permeabilities for concrete and laboratory-compacted clay. The lengths of a uniform plug required to limit seepage to 1 m^3/yr for varying depths of the disturbed rock zone (DRZ) are shown in Fig. 2 for a plug permeability of 10^{-9} cm/s.

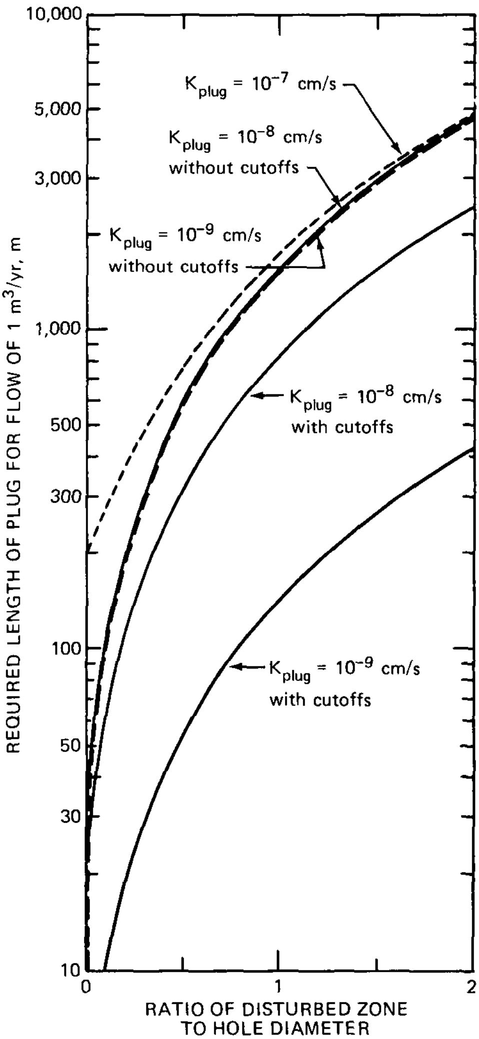

Figure 2 Plug length calculated from idealized numerical model.

Computations show that the length of plug necessary to meet the criteria of 1 m^3/yr, assuming no DRZ surrounding the plug, could be as little as 2 m for a plug permeability of 10^{-9} cm/s and 20 m for a plug permeability of 10^{-8} cm/s. The computations also show that flow through a plug of either permeability is minimal compared with that through the disturbed basalt. For a disturbed zone equal to the diameter of the plug (e.g., 21 m OD boundary) and a permeability of 10^{-7} cm/s, a plug about 1600 m long would be necessary.

These results indicate the importance of controlling the permeability of the disturbed zone and the inefficiency of requiring that the permeability of the plug material be more than an order of magnitude less than can be expected for the rock surrounding the plug. One way to reduce flow through a disturbed rock zone may be by installing cutoffs. These are 1- or 2-m-wide excavations through the disturbed rock zone surrounding a shaft or tunnel excavation and are repeated at intervals along the plug length, so that total length of cutoffs approximates about 10% of plug length. Figure 2 shows the results of computations evaluating the effectiveness of cutoffs. For a disturbed zone equal to the plug diameter, the necessary lengths of plugs with cutoffs are reduced to approximately 150 and 840 m for the two permeabilities studied, respectively.

Current thinking is that significant permeability increases might be found only within 2 m or less of the excavation wall and that more definitive data may be found in: (1) joint permeability experiments (Iwai, 1976), (2) cleavage cracking phenomena near an excavation wall (Rey, 1981) relative to the presence of stress loads implied by the occurrence of core disking, reported in field exploration work at Hanford (Myers and Price, 1979), and (3) field experiments currently under way at the Colorado School of Mines experimental mine facility, which involve drill-hole packer tests of permeability behind controlled “drill and blast” adit walls.

As summarized in Fig. 2, the thickness and permeability of the disturbed zone are the major factors that influence the amount of flow. The results presented are based on a tunnel diameter of 7 m.

Radionuclide Barrier Analysis

Radionuclide barrier performance of the plug was analyzed numerically to estimate the minimum length of plug required to keep the maximum credible concentration of specific radionuclides at the end of the plug below the maximum allowable concentration stipulated by Nuclear Regulatory Commission (NRC) draft regulation (Nuclear Regulatory Commission, 1981). At each step of the numerical analysis, numerous uncertainties or unknowns had to be covered by conservative assumptions. Idealized closed-form solutions were used to obtain first-approximation data. The following methodology was used:

- Maximum credible concentration in the repository was estimated for the assumed most critical transuranic and nontransuranic isotopes.
- Radionuclide migration through the plug was modeled numerically according to the theory of mass transport through a saturated porous medium.

In this preconceptual analysis, radionuclides were selected by considering the half-life of radioisotopes and choosing those with potentially more critical concentrations at the end of plugs. Table 1 lists the radionuclides chosen and summarizes decay characteristics and concentration limits for the analysis.

To estimate the quantity of radionuclides escaping from the canisters, we made the following assumptions:

1. Each canister contains three waste fuel assemblies.
2. The canisters are cooled for a 10-yr period before storing in a repository.
3. When a waste package fails, the total amount of radionuclides present in the canister is dissolved in the groundwater.

Table 1 Radionuclides Chosen in Numerical Analysis

Isotope	Maximum allowable concentration,* mμCi/ml	Half-life, yr
^{14}C	8×10^{-4}	5.73×10^{3}
^{129}I	6×10^{-8}	1.57×10^{7}
^{238}U	4×10^{-5}	4.51×10^{9}
^{239}Pu	5×10^{-6}	2.44×10^{4}
^{241}Am	4×10^{-6}	4.58×10^{2}
^{237}Np	3×10^{-6}	2.16×10^{6}

**Values from 10CFR Part 20, Appendix B, Table II, Column 2 (Nuclear Regulatory Commission, 1981). Concentration limits are within the boundary of the restricted area.*

4. The surrounding basalt rock is a no-flow boundary.
5. The volume of water per failed canister in which radionuclides are dissolved is computed by assuming a spacing of 15 m between waste packages and equals the amount of groundwater necessary to fill the void of a 15-m-long by 7-m-diameter repository section, with the backfill material between canisters having an effective porosity of 0.33 (e.g., 1.92×10^8 ml).

Assumptions 3 and 5 are believed to be highly conservative, and concentrations of listed radionuclides are expected to be less than the corresponding maximum credible concentration summaries provided in Table 2.

The theoretical approach used to model the migration of dissolved radioisotopes through a saturated porous medium was developed by Aikens et al. (1979) for shallow land-burial trenches. The model includes the reduction of radionuclide concentration by dispersion, sorption, and decay of radioactive isotopes in space and time. In this approach the movement of the dissolved ions through the

Table 2 Maximum Credible Concentration Within the Repository

Isotope	Concentration at 10-yr cooling time, Ci/spent fuel assembly	Maximum credible concentration in repository, mμCi/ml
^{14}C	2.5 E − 1*	3.91 E − 3
^{129}I	8.4 E − 3*	1.31 E − 4
^{238}U	1 E − 5†	1.56 E − 3
^{239}Pu	5.2 E − 4*	8.11 E − 2
^{224}Am	1.2 E − 3*	1.87 E − 1
^{237}Np	4.3 E − 1*	6.70 E − 1

**From Battelle Pacific Northwest Labs. (1976), Table 2.4.*
†Calculation made assuming 3×10^5 g per assembly.

subsoil is described by three hydraulic parameters:

1. Hydraulic velocity, which describes the movement of transporting groundwater through the subsoil
2. Dispersion coefficient, which describes the movement of radionuclides caused by the spatial gradient of radionuclide concentration
3. Retardation coefficient, which describes the ability of the subsoil to impede by sorption the movement of a specific radionuclide

Assumptions used to approximate the complex problem of the migration of dissolved radionuclides through the plug and its environment comprise a one-dimensional problem, modeled by one-dimensional, half-space medium, mass transport equations. The closed-form analytical solutions for the equations use the following additional radiological, thermomechanical, and hydrological design assumptions:

- The initial radioactivity of the contents of the repository decays with time and is constantly depleted by dissolution of radionuclides in the groundwater and by migration through the plug and out of the repository. These phenomena affect the concentration of contaminants by defining a time-dependent behavior of the source term. For this study, however, and because closed-form analytical solutions can be obtained only when steady-state conditions are reached, a constant source term was used. Thus the quantity of radionuclides available for migration has been conservatively assumed to be constant with time.
- The concentration of contaminant at the beginning of the plug was assumed to be the same as that in the repository near the waste canister; thus dispersion and sorption phenomena were neglected in the repository.
- The interaction behavior of radionuclides was neglected; thus the sorption capability of the medium for one specific radionuclide is independent of the concentration of the other radionuclides.
- The groundwater is incompressible and remains at constant viscosity in space and time.
- The thermal loading generated by the activity of the contaminants has not been taken into consideration in this study, although hydrogeological parameters and plug characteristics are sensitive to temperature change and temperature gradient.
- The plug is saturated.

With these assumptions, the mass transport equations give a relation between the length of plug required so that the maximum credible concentration of each specific radionuclide at the end of the plug is equal to the concentration limit defined by NRC regulations and the hydrological and radiological design parameters. This relation is defined by:

$$\ell = \frac{2\alpha}{\Theta} \ln \frac{c_0}{c_\ell} \tag{2}$$

where ℓ = length of the plug
α = longitudinal dispersivity coefficient
c_0 = concentration of the radionuclide at the plug face
c_ℓ = concentration limit of the radionuclide

$$\Theta = [1+(4\lambda DR/V)]^{1/2} - 1 \tag{3}$$

where λ = decay constant of the radionuclide
D = dispersion coefficient, $\simeq \alpha V$
R = retardation coefficient
V = hydraulic velocity of the groundwater
α = longitudinal dispersivity of the porous medium

Half-life of radionuclide, initial concentration at the plug face, and concentration limit values are presented in Tables 1 and 2 for the selected radioisotope.

If the plug is long enough in comparison with its transverse section, the average hydraulic velocity of the groundwater in the plug can be determined by Darcy's law:

$$V = \frac{1}{n} K_i \tag{4}$$

where V = hydraulic velocity (cm/s)
n = effective porosity
K = permeability (cm/s)
i = hydraulic gradient

Laboratory tests were conducted to provide quantitative values of the porosity and permeability of preferred candidate plug materials. For soil backfill and concrete plugs, porosities of 0.2 to 0.4 and permeability coefficients of 10^{-8} to 10^{-9} cm/s have been reported. Preliminary analyses also showed the extreme importance of the permeability of the disturbed zone in the plug-host-rock seepage analysis. In this study the average values chosen were ⅓ for the effective porosity and 10^{-7} cm/s for the hydraulic conductivity coefficient. For ^{129}I plug length will also be calculated for hydraulic conductivity of 10^{-8} cm/s to evaluate the sensitivity of plug length to a decrease in permeability. In all the computations, the space- and time-averaged hydraulic gradient is 10^{-3}. This value was proposed as a preconceptual datum for the long-term horizontal hydraulic gradient in the basalt repository.

The dispersion coefficient is a measure of the movement of the radionuclide in relation to the movement of the groundwater. The dispersion of the contaminant is caused principally by molecular diffusion and hydrodynamic dispersion (Nuclear Regulatory Commission, 1979). Molecular diffusion takes place when an unequal distribution of contaminant exists in a limited volume of water. This parameter has been neglected in this study. Hydrodynamic dispersion is caused principally by varying velocity in the groundwater system and can be approximated by

$$D = \alpha V \tag{5}$$

where V is the hydraulic velocity of the groundwater and α is the

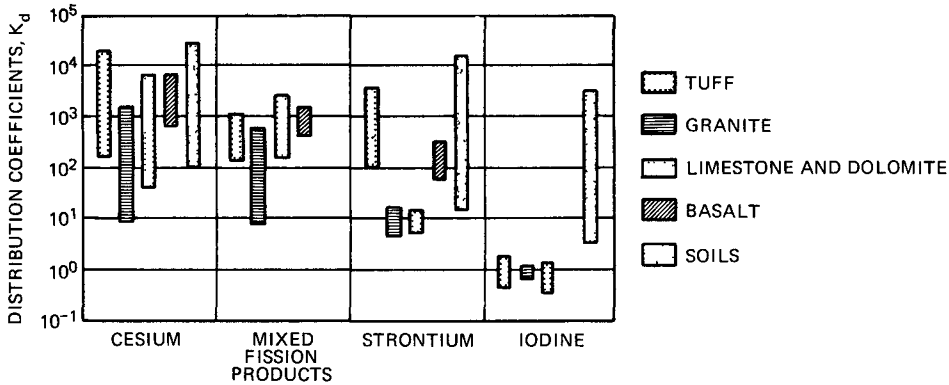

Figure 3 Ranges of distribution coefficients for various rock types. (Data from Grove, 1970.)

longitudinal dispersivity of the porous medium.

In the absence of laboratory test data for specific preferred plug material, a mean value of 3.048 m was chosen, based on data from the literature (Aikens et al., 1979; Nuclear Regulatory Commission, 1979). Retardation of radionuclide migration through a porous medium means that the rate of migration of the radionuclide is slowed to less than the rate of the groundwater. Retardation is caused principally by sorption phenomena between the contaminant and the porous medium. The retardation coefficient, R, depends on the distribution coefficient, K_d:

$$R = 1 + (1 - n)\, K_d/n \qquad (6)$$

where n is the effective porosity of the porous medium and K_d can be approximated by:

$$K_d = \frac{A}{100} \times \frac{Q}{Ce} \qquad (7)$$

where A is the dimensionless cation sorption equilibrium constant, Q is the cation exchange capacity (meq/100 g), and Ce is the total cation concentration (meq/ml).

Distribution coefficients are strongly dependent on the chemical nature of the radionuclide. Estimated values from Aikens et al. (1979) for a typical desert soil indicate, for example, a difference of four orders of magnitude between distribution coefficients of iodine and plutonium. The distribution coefficients are equally affected by the sorptive capacity of the medium, as shown in Fig. 3, where a range of values of the distribution coefficient is given for various rock types and radioisotopes. Distribution coefficients for iodine are reported to vary from 1 for granite or limestone to 5000 for a highly sorptive soil. For a specific radionuclide, the highest retardation coefficient can be expected for backfill (including clay montmorillonite or zeolite, considering their high exchange capacity). As stated by Schneider and Platt (1974), however, a side effect of sorption is that exchangeable ions are released from the medium into solution and can compete with the waste material for available exchange capacity. Values of retardation coefficients corresponding to those estimated by Aikens et al. for typical desert soil (see Table 3) are used in the following numerical analysis. They are conservative for a soil plug, and no

Table 3 Estimated Distribution Coefficients in a Typical Desert Soil*

Atomic number	Element	K_d,† ml/g	1/R‡
1	Tritium	0	1
4	Beryllium	75	3×10^{-3}
6	Carbon	2	1×10^{-1}
11	Sodium	10	2×10^{-2}
17	Chlorine	0	1
18	Argon	0	1
19	Potassium	35	6×10^{-3}
20	Calcium	15	1×10^{-2}
26	Iron	150	3×10^{-4}
27	Cobalt	75	3×10^{-3}
28	Nickel	80	3×10^{-3}
34	Selenium	20	3×10^{-2}
36	Krypton	0	1
37	Rubidium	125	2×10^{-3}
38	Strontium	20	1×10^{-2}
39	Yttrium	2,000	1×10^{-4}
40	Zirconium	2,000	1×10^{-4}
41	Niobium	2,000	1×10^{-4}
42	Molybdenum	5	4×10^{-2}
43	Technetium	0	1
46	Palladium	250	9×10^{-4}
48	Cadmium	2,000	1×10^{-4}
50	Tin	250	9×10^{-4}
51	Antimony	15	1×10^{-2}
53	Iodine	0	1
55	Cesium	200	1×10^{-3}
61	Promethium	600	4×10^{-4}
62	Samarium	600	4×10^{-4}
63	Europium	600	4×10^{-4}
67	Holmium	600	4×10^{-4}
81	Thallium	2	1×10^{-1}
82	Lead	4,000	6×10^{-5}
83	Bismuth	10	2×10^{-2}
84	Polonium	25	9×10^{-3}
85	Astatine	0	1
86	Radon	0	1
87	Francium	200	1×10^{-3}
88	Radium	100	2×10^{-3}
89	Actinium	1,000	2×10^{-4}
90	Thorium	15,000	2×10^{-5}
91	Protactinium	4,000	6×10^{-5}
92	Uranium	3,000	7×10^{-5}
93	Neptunium	15	1×10^{-2}
94	Plutonium	2,000	1×10^{-4}
95	Americium	2,000	1×10^{-4}
96	Curium	600	3×10^{-4}
97	Berkelium	700	3×10^{-4}

**Data from Aikens et al., 1979.*
†Equilibrium distribution coefficients between water and soil.
‡Inverse of retardation coefficient.

published data were available for a concrete material. These values are expected to provide a first estimate of the retardation phenomena in a multizone soil-concrete plug. Plug length is also computed for ^{129}I, with 10 as an average value of the plug retardation coefficient, to simulate a sorptive montmorillonite-zeolite backfill plug.

The minimum plug lengths required to keep the concentration of contaminant at the end of the plug below the limits of the NRC draft (Nuclear Regulatory Commission, 1981), assuming initial maximum credible concentrations from Table 2 and retardation coefficients from Table 3, are:

Element	Length, m
^{14}C	0.8
^{239}Pu	0.6
^{224}Am	0.1
^{235}U	168.3
^{237}Np	194.6
^{129}I	16,507.0

The estimated minimum required plug lengths are very small for carbon, plutonium, and americium, significant for uranium and neptunium, and extreme for iodine.

A sensitivity study was done on uranium and neptunium to estimate the variation in required plug length in relation to the concentration in the repository. Curves given in Fig. 4 for neptunium and uranium show that, for an increase of two orders of magnitude in the concentration of either uranium or neptunium, a plug only twice as long is required. In addition, because permeability and retardation coefficients used in this analysis are believed to be conservative for plug material performance, the minimum lengths shown in Fig. 4 should give an acceptable upper limit for preconceptual plug design for uranium or neptunium.

For iodine the numerical modeling gives an estimated plug length of 16 km. As shown in Fig. 5, curve A, even if the concentration in the repository is three orders of magnitude less than the maximum credible concentration assumed, the required plug length is still 4 km. Therefore a higher performance plug is required. Two potential ways to improve plug performance are to decrease the permeability and to increase the retardation characteristics of the plug.

The analytical formulation of the required plug length given by Eq. 2 indicates that equivalent plug length is influenced by increasing the retardation coefficient or by decreasing the permeability in the same order of magnitude. For iodine the numerical analysis shows that plug length is directly proportional to an increase in the retardation coefficient or a decrease in the permeability coefficient.

The computations for iodine, which led to a 16-km-long plug, were for a nonsorptive material with respect to iodine ($R = 1$) and a relatively permeable plug ($K = 10^{-7}$ cm/s). On the basis of test data for plug material permeability and the range of distribution coefficients given by Grove (1970) for iodine in soils (see Fig. 3), it seems feasible to construct a more efficient plug.

Curves B and C in Fig. 5 indicate that: (1) For a retardation coefficient $R = 10$ (alternately, $R = 1$) and a plug hydraulic conductivity $K = 10^{-7}$ cm/s (alternately $K = 10^{-8}$ cm/s), the estimated plug length is 1620 m. (2) For a retardation coefficient $R = 100$ (alternately $R = 10$) and a plug hydraulic conductivity $K = 10^{-7}$ cm/s (alternately, $K = 10^{-8}$ cm/s), the required plug length is less than 200 m.

In summary, the equal and direct dependence of plug performance on permeability and sorptive ability provides guidelines to optimize plug

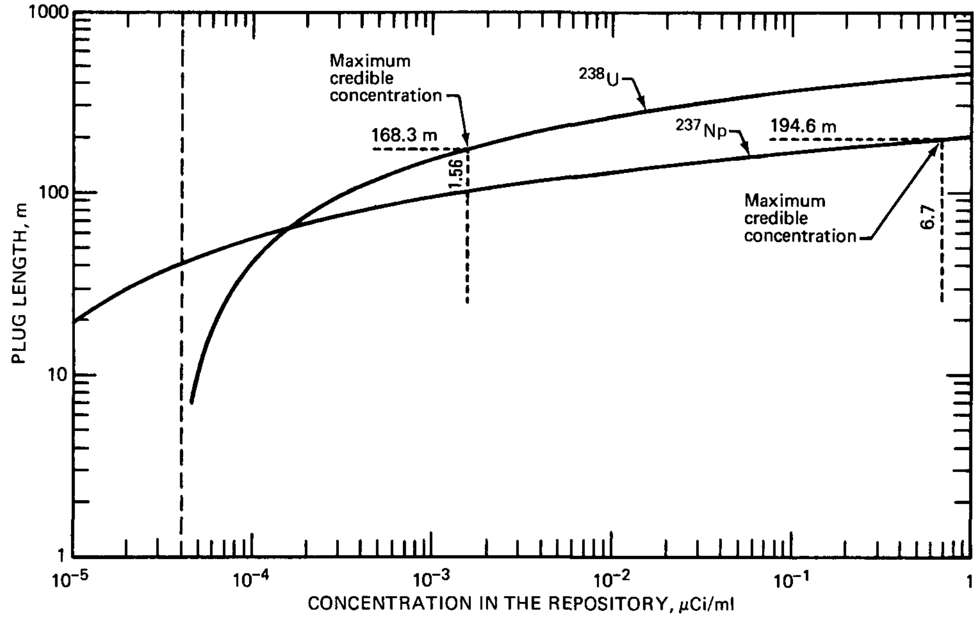

Figure 4 Plug lengths required (Nuclear Regulatory Commission, 1981) for concentration limits at end of plug. Limit for ^{238}U is indicated by dashed vertical line. Limit for ^{237}Np is 3×10^{-6} μCi/ml.

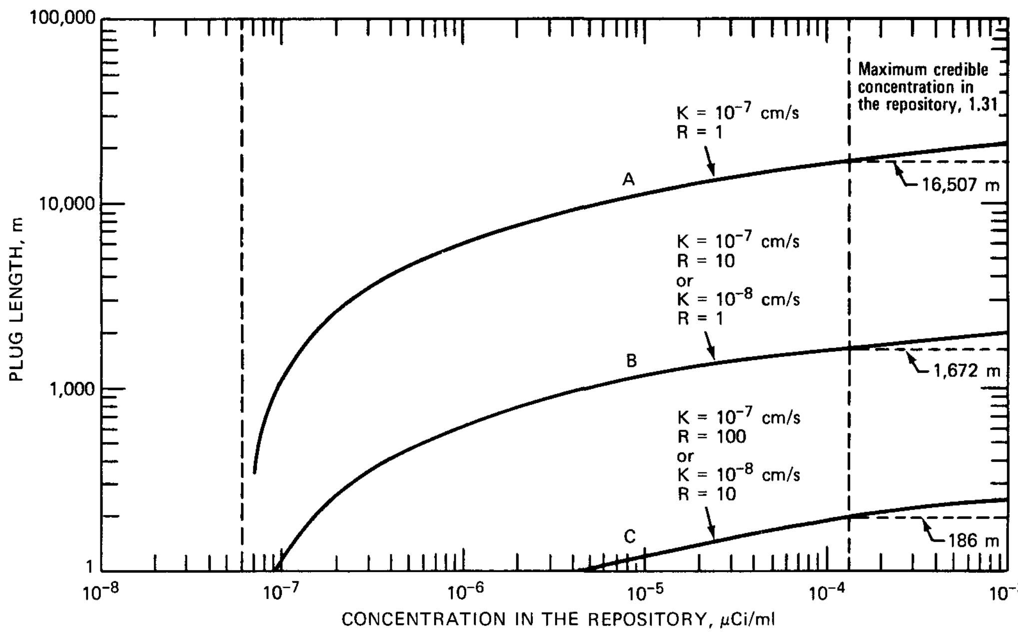

Figure 5 Plug lengths required (Nuclear Regulatory Commission, 1981) for concentration limits at end of plug and sensitivity study for ^{129}I with respect to K and R. Limit for ^{129}I is indicated by dashed vertical line.

design. The sensitivity analyses also indicated that quality control of seal construction is of primary importance to ensure acceptable plug performance.

Monolithic Plugs and Performance

After the dimensional bounds and performance ranges of the plug as a seepage control and radionuclide barrier were analyzed, the next step was to identify engineering performance categories and criteria that could be used to rate candidate plug designs. Figure 6 shows the selected performance categories, identified in the figure as "Plug Design Functions." All five of the functions are essential to plug performance in large-diameter shafts and tunnels; whereas only three are currently considered essential to small-diameter boreholes.

The design performance criteria for each plug design function are shown in Tables 4 through 6, together with a scale of numerical ratings to cover a potential range of performance for any candidate plug design relative to each criterion.

Candidate plugs were initially designed as monolithic plugs; i.e., the entire plug was designed as either compacted earth or concrete, or some other single material. Engineering evaluation of the monolithic plugs for performance in each of the plug design functions was then carried out.

The selection of preferred schemes for a preconceptual design involves at least two problems at this point:

1. Which scheme is best for a particular design function for a shaft?
2. Which scheme is best for the complete range of design functions for a shaft?

The first problem arises, for example, when we attempt to compare two competing schemes for the design function of "core barrier performance" in shafts and note that one scheme performs better for the parameter of "permeability" and another performs better for the parameter of "ion exchange." A quantitative measure of the relative importance of one parameter in a design function cannot be precisely known at this point. If a specific weighting formula could be devised, we might obtain a single weighted average value for all the parameters of a design function for any particular scheme to be compared with the weighted average obtained for other schemes. Nonetheless, design intent for the various schemes implicitly assumes certain inequalities among the parameters. For example, in the design function of core barrier performance, the following inequalities were assumed for the relative importance of parameters:

Permeability > cost
Permeability > ion exchange

Extended dominance analysis can be used to sort out the task of comparing alternate schemes once such inequalities are identified.

Extended dominance analysis was used to compare and identify superior schemes after they were rated according to the numerical criteria described in Tables 4 through 6. Dominance is one of the fundamental concepts of decision analysis. Simply stated, an alternative A is said to dominate alternative B if it can be shown that A is at least as desirable as B with respect to all evaluation measures (or geotechnical and chemical tests of preferred plug materials and trial performance criteria chosen during the study). The performance criteria were selected to be conservative in nature, and simplified, closed-form solutions were used in the analysis of performance. Considerably more work needs to be done to verify the design assumptions, both analytically and by scaled and prototype field tests.

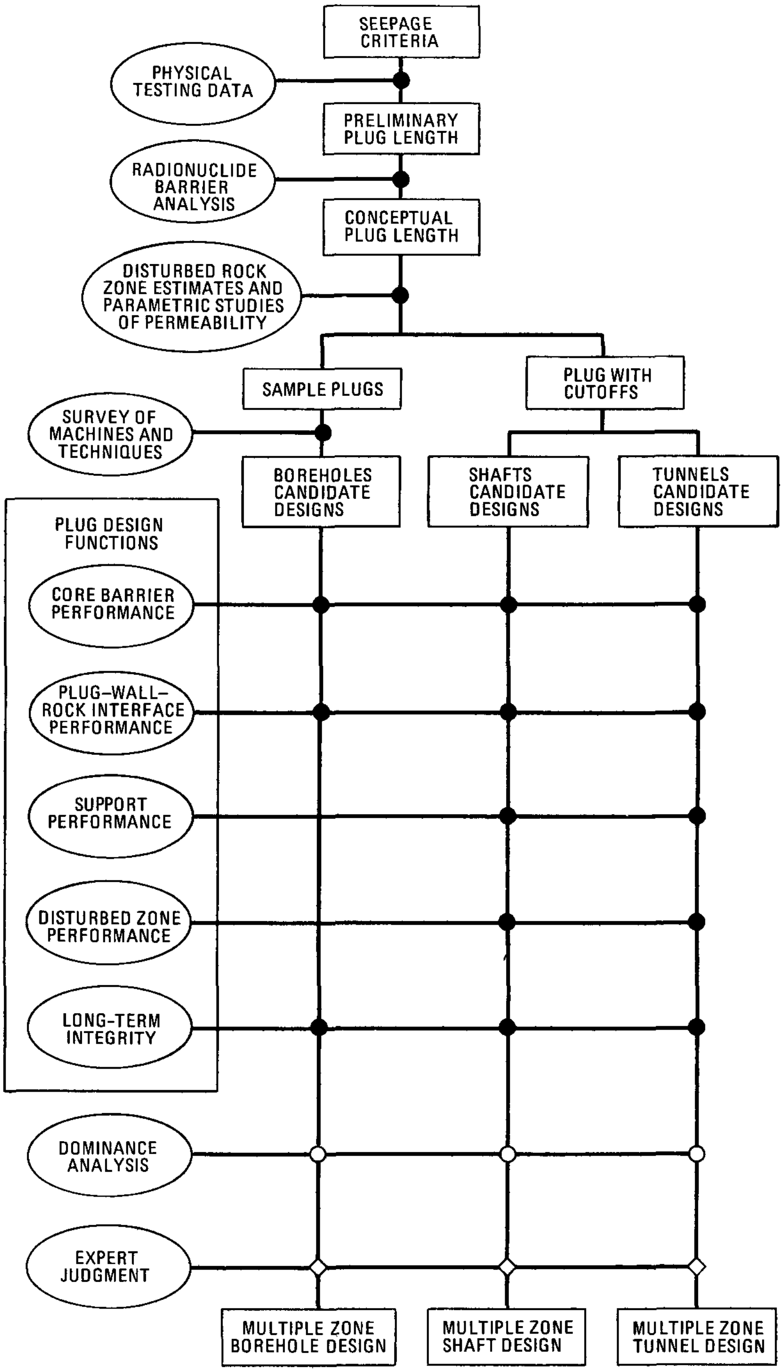

Figure 6 Task flow diagram for plug designs.

Table 4 Plug Design Function Evaluation Criteria: Plug and Disturbed Zone Performance

Parameter	Scale	Description
Plug and Disturbed Zone Mechanical Stability Performance		
Plug mechanical stability under in-situ thermomechanical loading	0	No stability
	1	Stable when plug confined (no axial movement)
	2	Stable even when plug is not confined
Long-term wall rock support	0	No long-term support provided
	1	Long-term support provided by plug strength resistance *or* by expansive properties of plug
	2	Long-term support provided by plug strength resistance *and* expansive property of plug
Disturbed Zone Performance		
Disturbed zone treatment performance	0	Cutoff collars do not improve disturbed zone treatment better than contact grouting ($K \geqslant 10^{-6}$ cm/s)
	1	Cutoff collars, permeability $\sim 10^{-7}$ cm/s
	2	Cutoff collars, permeability $\sim 10^{-8}$ cm/s
	3	Cutoff collars, permeability $\sim 10^{-9}$ cm/s
Construction difficulty	0	Very difficult
	1	Moderate difficulty
	2	No special difficulty

Table 5 Plug Design Function Evaluation Criteria: Plug–Wall-Rock Interface and Barrier Performance

Parameter	Scale	Description
Plug–Wall-Rock Interface Performance		
Sliding stability	0	Sliding stability nonexistent or has to be demonstrated
	1	Acceptable long-term *or* short-term sliding stability
	2	Acceptable long-term *and* short-term sliding stability
Plug-wall-rock continuity after cyclic thermo-mechanical loading	0	No continuity
	1	Moderate continuity
	2	Expected continuity if plug confined
	3	Good continuity
Reliability of interface joint closure during construction	0	Poor reliability
	1	Moderate reliability
	2	Good reliability
	3	Good reliability and self-healing potential
Plug Barrier Performance		
Ion-exchange capacity (absorptive capability of plug)	0	Low (5 to 40 meq/100 g)
	1	Moderate (40 to 80 meq/100 g)
	2	High (80 meq/100 g)
Permeability of plug material as determined in laboratory tests (K, cm/s)	1	$10^{-7} \leqslant K < 10^{-8}$
	2	$10^{-8} \leqslant K < 10^{-9}$
	3	$K < 10^{-9}$
Uniformity of permeability through plug	0	2 orders of magnitude
	1	1 order of magnitude
	2	Uniform
Unit cost of 100-m-long plug	0	Research and development: no price given
	1	Price > $1,000,000
	2	Price < $1,000,000

Table 6 Plug Design Function Evaluation Criteria: Plug Long-Term Integrity Performance

Parameter	Scale	Description
Solubility	0	Affected by pH 7 to 10
	2	Stable in pH 7 to 10
Documented history of survival	0	Less than 2000 yr of relevant documented history of survival with no analogs of similar material in nature
	1	Man-made material; an analog or the actual material can be documented for 2000 yr
	2	Natural material; survival history of millions of years in nature
E_h	1	Stable under oxidizing conditions (E_h of environment, 0.0 volts)
	2	Stable under reducing conditions (E_h of environment, 0.0 volts)
	3	Stable through a range of E_h conditions from moderately oxidizing to reducing

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Nuclear Waste Repository in Basalt: A Design Description

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The conceptual design of a nuclear waste repository in basalt is described. Nuclear waste packages are placed in holes drilled into the floor of tunnels at a depth of 3700 ft. About 100 miles of tunnels are required to receive 35,000 packages. Five shafts bring waste packages, ventilation air, excavated rock, personnel, material, and services to and from the subsurface. The most important surface facility is the waste handling building, located over the waste handling shaft, where waste is received and packaged for storage. Two independent ventilation systems are provided to avoid potential contamination of spaces that do not contain nuclear waste. Because of the high temperatures at depth, an elaborate air chilling system is provided. Because the waste packages deliver a considerable amount of heat energy to the rock mass, particular attention is paid to heat transfer and thermal stress studies.

Introduction

Over the last 35 yr, nuclear power reactors, the defense program, and other users of radioactive materials have produced large quantities of nuclear waste. This waste is now in temporary storage, and, sooner or later, it must be disposed of permanently. More nuclear waste is being produced, but, even if all waste production were ended today, the existing waste would still have to be managed. Fortunately solutions to the nuclear waste problem are at hand. Safe disposal in deep, impervious geologic strata is feasible, and work is under way to enable the construction of such nuclear waste repositories in one or several locations in various geologic media. This paper describes the conceptual design of a nuclear waste repository in hard rock, namely, basalt.

A nuclear waste repository in hard rock should be designed to:

- Provide and preserve multiple, natural, and man-made barriers to protect against unacceptable radionuclide migration
- Ensure retrievability without damage to waste packages
- Limit rock movements that may cause permeability enhancement

The conceptual design of the Nuclear Waste Repository in Basalt (NWRB) was started by Kaiser Engineers, Inc., and Parsons Brinckerhoff Quade & Douglas, Inc., in October 1979, under contract with the Department of Energy, Richland Operations Office and under the technical direction of Rockwell Hanford Operations, Office of Repository Studies.

Figure 1 is an artist's conception of the repository showing the extent of the underground facilities and the five access shafts connecting the subsurface with the surface facilities above. The shaft pillar is approximately in the center of the subsurface facilities.

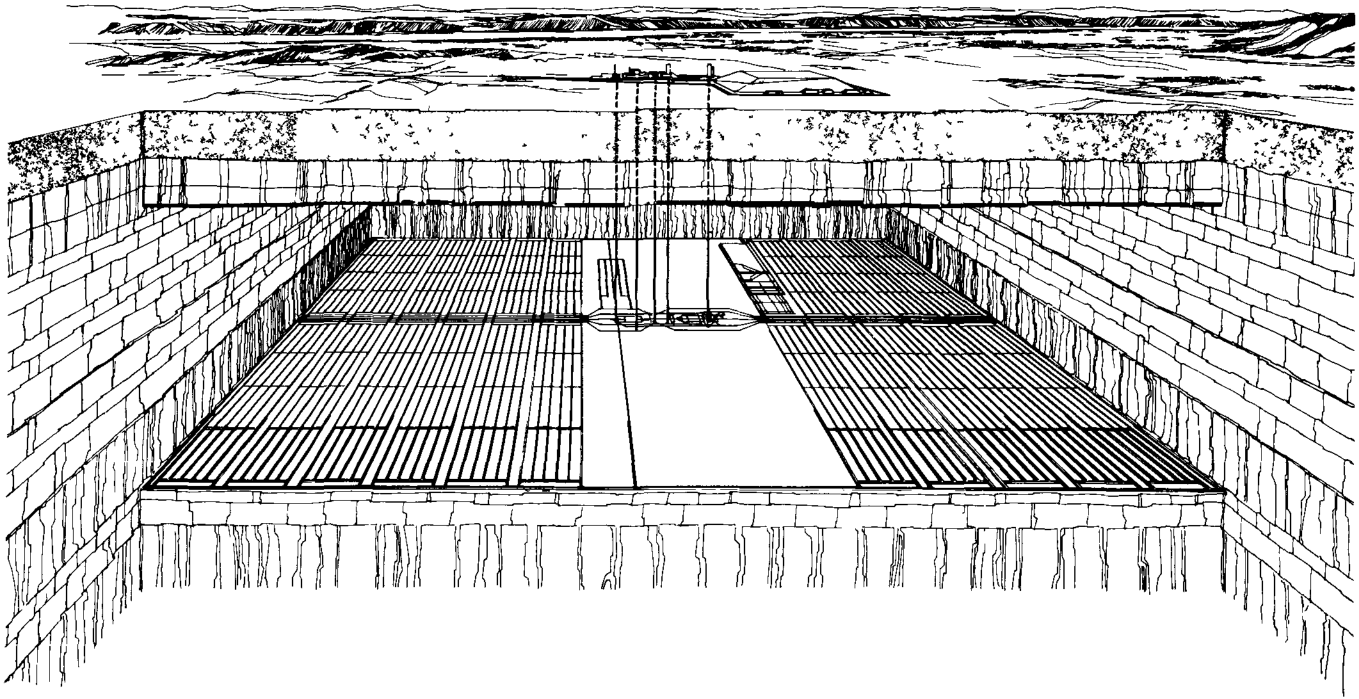

Figure 1 Mine cutaway and surface perspective.

Nuclear waste (spent fuel) is received at the surface storage facilities in canisters transported in a shipping cask. The canisters are unloaded and overpacked in a waste handling building to form waste packages, which are then lowered to the storage horizon, transported to vertical storage holes in the storage rooms, and lowered into them.

The NWRB is intended to receive canisters of disassembled spent fuel assemblies and drums of low-level waste (LLW) and to overpack and store the spent fuel in an engineered barrier configuration. Waste will be stored so that it is retrievable for 25 yr. The NWRB is to be licensed by the Nuclear Regulatory Commission.

For design purposes, spent fuel is considered to be 10 yr old at the time of storage. The design provides for receipt of 1750 canisters per year for 20 yr. Each canister contains the disassembled rods from three pressurized water reactor (PWR) or seven boiling water reactor (BWR) fuel assemblies. The ratio is 37% PWR to 63% BWR fuel assemblies. Total capacity is 47,000 tonnes of heavy metal. Expansion capability is provided in the repository. The design rate for LLW is 1600 drums per year for 20 yr.

Principal Design Assumptions. Principal design assumptions for the waste form and waste package are:

- Each canister contains rods from three PWR fuel elements or equivalent.
- Fuel is 10 yr old when received.
- Canisters are overpacked with buffer and barrier at the repository.
- The repository is to be designed for 35,000 waste packages, 20 yr of waste placement of 1750 waste packages per year.
- Waste must be retrievable for 25 yr after emplacement.

Assumed properties of intact rock are:

- Density, 2.8 g/cm^3
- Uniaxial compressive strength, **200 MPa**
- Tensile strength, 14 MPa
- Young's modulus, 67,800 MPa
- Poisson's ratio, 0.26
- Thermal expansion coefficient, 7.45×10^{-6}°C
- Specific heat, 1.0 kJ/(kg · °C)
- Thermal conductivity, 1.56 W/(m · °C)
- Diffusivity, 6.5×10^{-7} m^2/s

Assumed conditions at repository depth are:

- Repository storage horizon depth, 1128 m
- Assumed ratio between horizontal and vertical stresses, 1.0
- Initial in situ rock temperature, 57°C
- Basalt flow thickness (Umtanum), 62.5 to 87 m
- Discontinuities (tight joints), 3 to 10/m

Principal Design Criteria. Maximum temperatures established as design criteria (figures actually achieved are shown in parentheses) are:

- Fuel cladding, 300°C (300°C)
- Basalt, 500°C (less than 200°C)
- Entries
  Operational phase, 70°C (~60°C)
  Retrievable storage phase, 100°C (~60°C)
  Terminal storage phase, 150°C (70°C)

Rock stress safety factors are shown in Table 1.

Reference Stratigraphy

The reference location of NWRB on the Hanford site is underlain by about 213 m of poorly consolidated granular overburden, followed by a thick sequence of basalt flows; at least 70

Table 1 Minimum Rock Stress Safety Factors*

	Phase		
Location	Operational	Retrievable storage	Terminal storage
Storage hole	1.5 (4.2)	1.0 (1.9)	1.0 (>1.9)
Storage room	2.0 (2.7)	1.25 (2.1)	1.0 (1.8)
Entries	2.0 (>2.7)	1.5 (>2.1)	1.0 (>1.8)

**Conceptual design figures achieved are shown in parenthesis.*

Table 2 Hanford Site Stratigraphy

Unit	Approximate thickness, m	Approximate number of flows
Hanford formation	0 to >100	
Ringold formation	0 to >200	
Interbeds (each)	0 to 10 m	
Saddle Mountains basalt	275*	10
Wanapum basalt	350*	11 to 14
Grande Ronde basalt	>2600	at least 50

**Includes interbeds.*

flows extend to a depth greater than 3.2 km. The upper 20 or so flows are interbedded with thin sedimentary units.

The basalt flows are divided into three formations; they are, from oldest to youngest: Grande Ronde basalt, at least 50 flows; Wanapum basalt, 11 to 14 flows; and Saddle Mountains basalt, approximately 10 flows. Many of the flows in the Wanapum and Saddle Mountains basalts are separated by sedimentary layers known as interbeds. Interbeds vary in thickness and are commonly laterally discontinuous, with an average thickness of about 10 m. The thickest interbeds occur in the Saddle Mountains basalt.

Overlying the basalts and their interbeds are two sedimentary formations—Ringold and Hanford formations. The Ringold formation consists of interbedded gravels, siltstones, and sandstones deposited by an extensive river-flood-plain system. The Hanford formation consists of unconsolidated gravels and sand deposited by catastrophic floods associated with continental glaciation. The stratigraphy is shown in Table 2.

A typical basalt flow has a low-density vesicular or brecciated flow top followed by a thick central entablature, which is a dense, impervious basalt fractured by cooling joints and secondary joints in varying patterns. Beneath the entablature is the colonnade, which is similar in density to the entablature but has predominantly vertical cooling joints in typical hexagonal patterns. Colonnade joints tend to be continuous; joints in the entablature are more likely to be discontinuous. Variations in this sequence are common, but the central portion of a flow is almost invariably dense, hard, and impervious. The present candidate repository basalt flow is the Umtanum flow at a depth of about 1128 m; it is between 62.5 and 87 m thick.

Access Shafts

The five access shafts, shown in order from left to right in Fig. 1, serve the five primary functions:

- Confinement air exhaust
- Nuclear waste hoisting
- Confinement air intake
- Personnel and materials hoisting and mine air intake
- Basalt hoisting and mine air exhaust

Ventilation is handled by two separate systems independent of each other; one serves the repository development (mining) and the other, areas containing nuclear waste (the confinement area).

In the conceptual design each shaft is sunk by conventional drill, blast, and muck techniques. The first 548 m [~213 m of upper sedimentary beds (Ringold and Hanford formations) and ~335 m of basalts with intermediary sedimentary beds], through the Priest Rapids basalt, will be frozen before sinking commences to control water inflow in the shaft area. The Priest Rapids basalt comprises the upper flows of Wanapum basalt.

Water-bearing zones expected to be encountered below the frozen section as sinking progresses will be grouted and tested before they are penetrated. Down to and through the Priest Rapids basalt, the shafts are designed to be lined with a steel cylinder embedded in concrete. Below the Priest Rapids, the lining will be iron tubbing backed by cast-in-place concrete. The steel cylinder and the tubbing will be sized to withstand hydrostatic pressures expected through individua water-bearing zones. The combination of grouting, concrete, steel cylinder, and tubing will be designed to limit water seepage down the shaft into the repository. Certain structural details needed to decommission the shafts will be constructed as the shafts are sunk. These details, which include proposed cutoff collars to retard potential radionuclide migration and water barriers to permit portions of the shaft lining to be removed during decommissioning, are shown in Fig. 2.

This sinking method was chosen for the conceptual design because it is a proven method for the shaft sizes (4 to 6 m excavated diameter through the rock) and depths under consideration. The freezing method for ground stabilization was developed and successfully applied in Europe in the early 1900s. In the 1960s and 1970s, several shafts were sunk by the freezing method to develop Canadian potash deposits, U. S. salt deposits, and German and British coal deposits, through unconsolidated, saturated sands. In Saskatchewan approximately 20 shafts were sunk through the brine-saturated Blairmore formation to depths approaching 600 m. Freezing has also been successfully used in Yorkshire, England, to a depth of about 915 m.

Cross sections of the five shafts are shown in Figs. 3 through 7. Shaft 1 exhausts air from the confinement ventilation system, i.e., from the waste storage areas. Shaft 2 is used for transporting nuclear waste to and from the underground facilities; it also exhausts a small portion of the confinement ventilation air. Shaft 3 supplies air to the confinement ventilation system. Shaft 4 is used for moving personnel and materials to and from the subsurface facilities and supplies air to the mine development ventilation system. Shaft 5 is used for transporting mined basalt to and from the underground facilities and also exhausts the mine development ventilation air. The shaft diameters, which range from 3.05 to 4.88 m, were minimized by using electric equipment in lieu of diesel, except for the waste transporter.

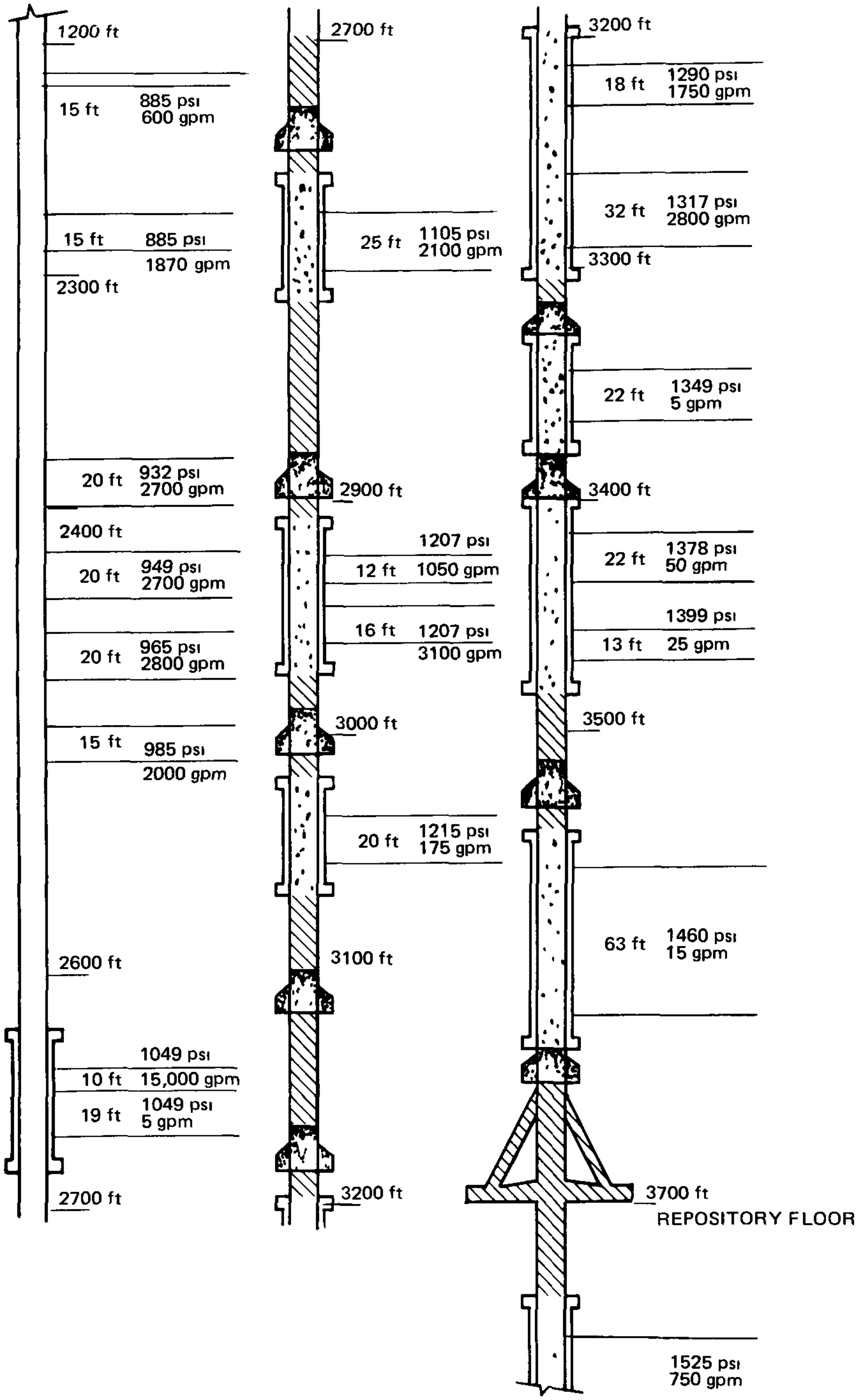

Figure 2 Waste handling shaft showing locations of cutoff collars and water barriers.

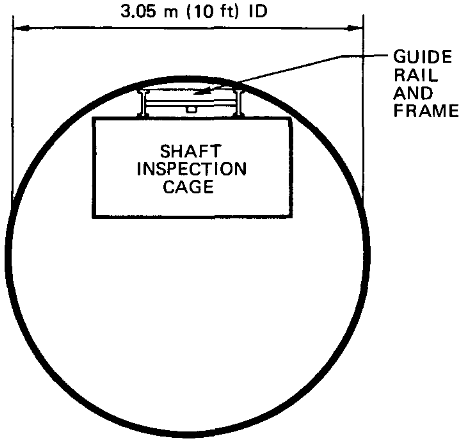

Figure 3 Confinement exhaust air shaft (shaft No. 1).

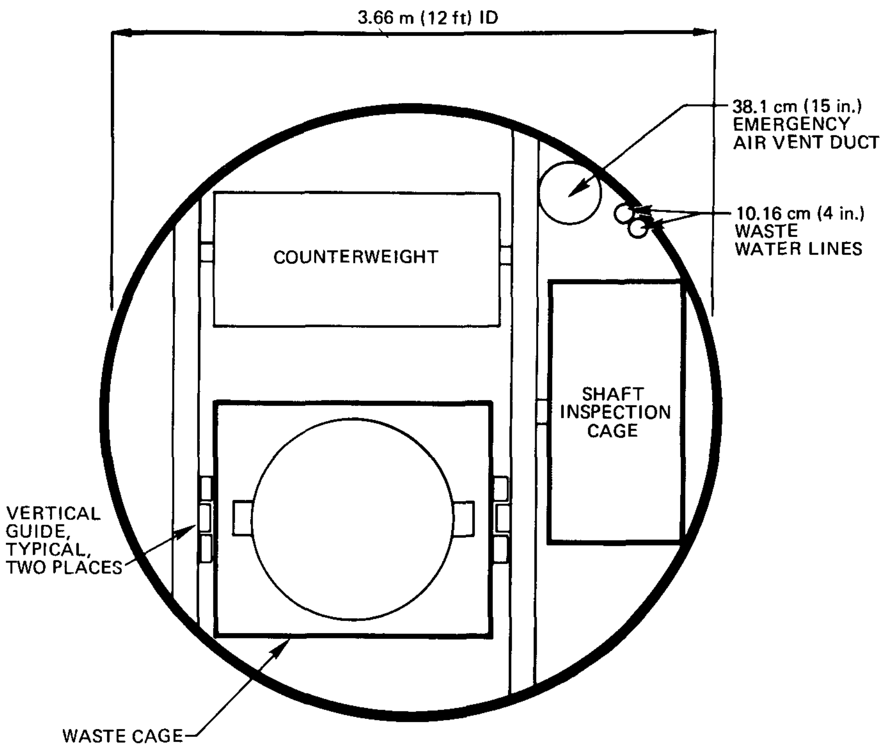

Figure 4 Waste transport shaft (shaft No. 2).

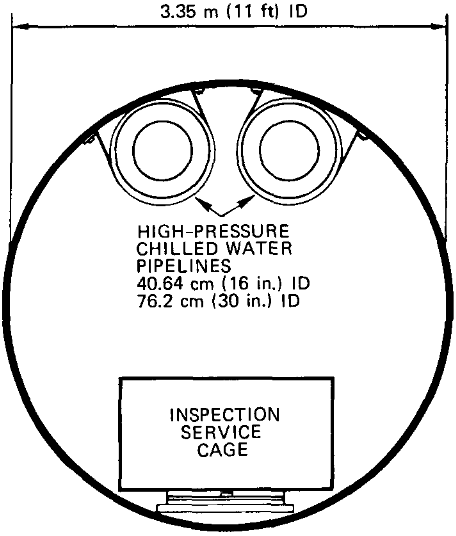

Figure 5 Confinement intake shaft (shaft No. 3).

Shaft Pillar and Ventilation

The shaft pillar area is extensive and complex (Fig. 8) because of the requirement to isolate the confinement and development areas from each other. The layout of the shaft pillar area (Fig. 9) identifies the shafts and the many operational functions that are provided in the shaft pillar development. Flow paths of the air through the shaft pillar are shown in Fig. 10 for confinement intake, confinement exhaust, mine development intake, and mine development exhaust ventilation.

The ventilation requirements for the repository are shown in Table 3. The independent air supply and return paths for the two separate ventilation systems are provided in four of the five repository shafts.

The ventilation system is not used to remove any significant portion of the natural heat from the radioactive waste. After waste is emplaced in the storage room, bulkheads are emplaced to prevent further airflow through the filled storage room except for a small airflow used for monitoring potential waste-package failures. When backfilling operations are initiated in a storage room, ventilation air will be used to cool the rooms sufficiently to permit such operations.

Because the ambient temperature of the rock at the 1128-m storage horizon will be ~57°C, intake air will require local cooling for the comfort of workers in specific areas. Cooling will be provided by local fan-coil units served by a chilled water system, which is described in the section entitled Thermomechanical Considerations.

Subsurface Layout

A plan of the subsurface facilities is shown in Fig. 11. The shaft pillar is near the center; to the north is the contact waste storage panel and to the northeast, an experimental panel. The remainder of the area is divided into 21 storage panels for spent fuel; one is a spare. The repository is 3207 m long by 2377 m wide.

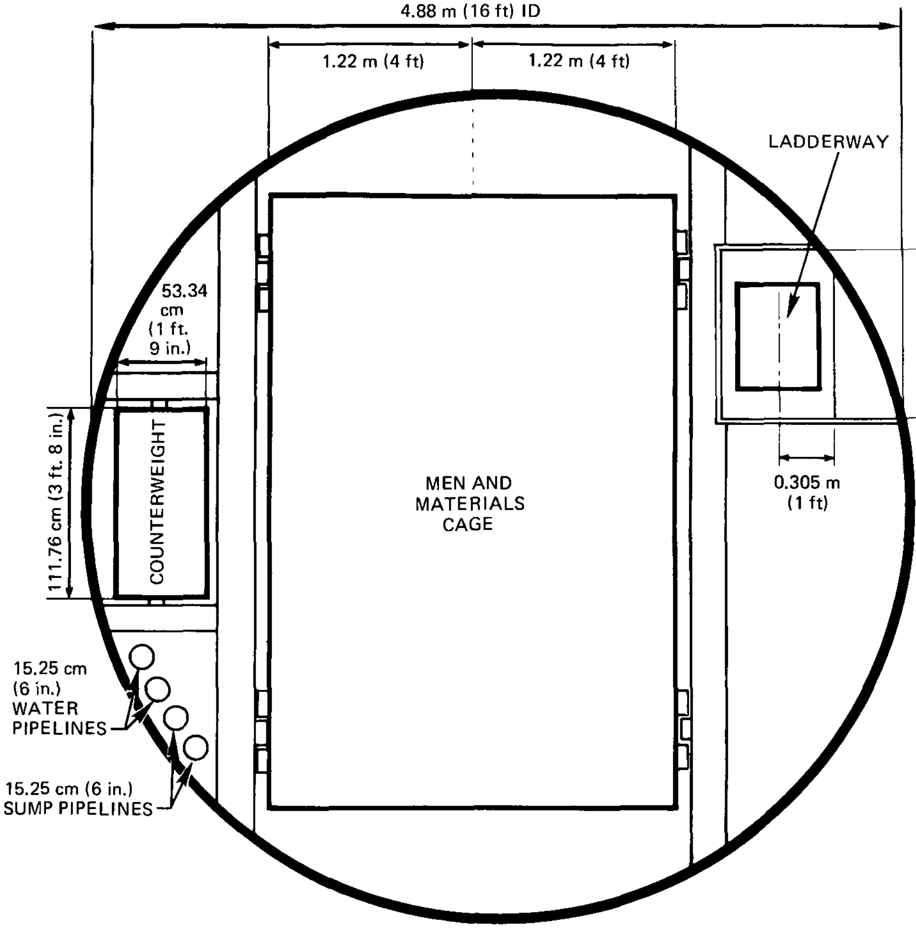

Figure 6 Service shaft, mine intake (shaft No. 4).

Five main entries parallel the long axis of the repository along the center line. One is for basalt hauling and mine air exhaust; two are for mine air intake, personnel, and materials; and two are for confinement air intake and nuclear waste hauling. The confinement exhaust airways are at the outer extremities of the repository.

The storage panels are rooms designed to store 1-yr's receipts, and each is isolated from adjacent panels. A typical storage panel for spent fuel is shown in Fig. 12. Each panel is 187 m wide by 1089 m long. Six storage rooms run the full length of each panel, with cross cuts at three intermediate locations for escape and ventilation purposes. The rooms are 4.3 m wide and are spaced 37 m from center to center, for an extraction ratio of about 8%.

Waste Storage

Figure 13 shows cross sections of the rooms and also the storage configuration within a room. An artist's view of a portion of a storage room in Fig. 14 shows typical waste emplacement vertically in the floor in holes drilled along the center line of the

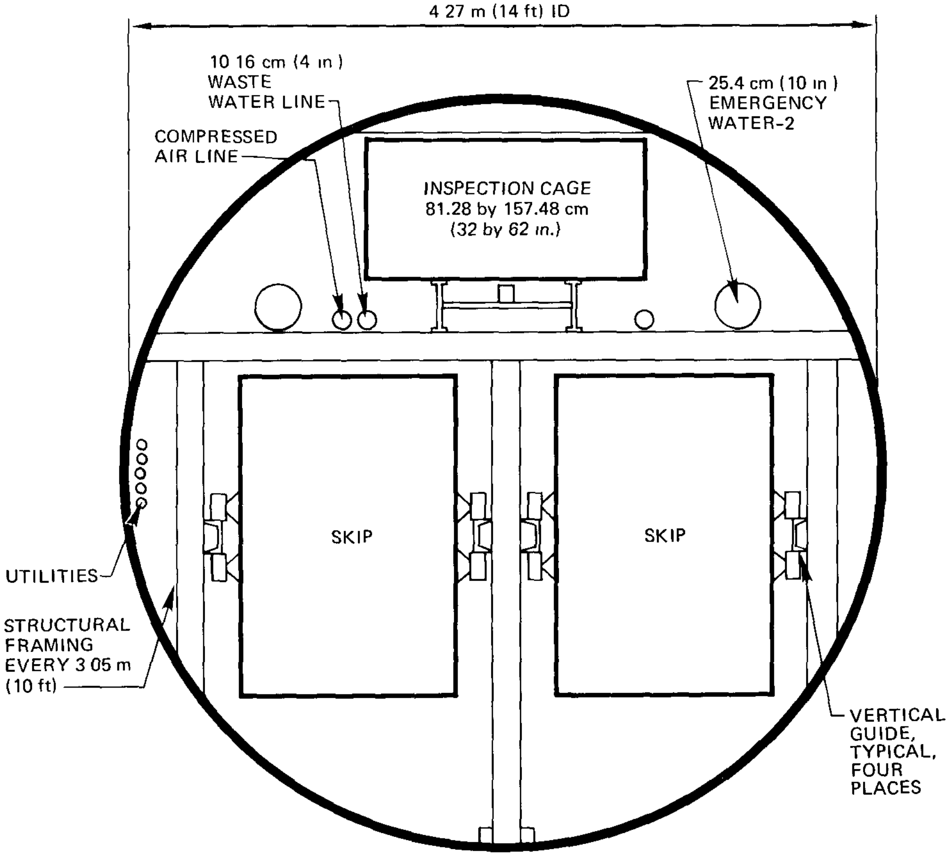

Figure 7 Basalt hoisting shaft, mine exhaust (shaft No. 5).

room. A typical storage position for spent fuel is shown in Fig. 15.

In the present concept a lining is placed before the waste package is stored; it consists of a compacted bentonite-crushed-rock mixture surrounding a mullite sleeve. The bentonite mix is enclosed in an aluminum container so that the sleeve can be prefabricated to allow easy installation. The entire sleeve unit is lowered into the hole and packed with a thin annulus of bentonite. A shield plug above the waste package decreases radiation levels to 2.5 mrem/hr.

Details of the spent fuel waste package are shown in Fig. 16. The canister received at the repository contains the disassembled spent fuel rods in the center of the waste package and is surrounded by a graphite buffer, part of the engineered barrier, which protects the carbon steel of the canister in case of water intrusion. The waste package has an outer shell of titanium into which the canister is sealed in the repository hot cell by laser welding.

Present plans require waste packages to be retrievable for 25 yr, a period of time long enough to demonstrate the safety of the repository and its components. The NWRB plans to keep storage rooms open but sealed for this period and to monitor the behavior of the waste packages, the

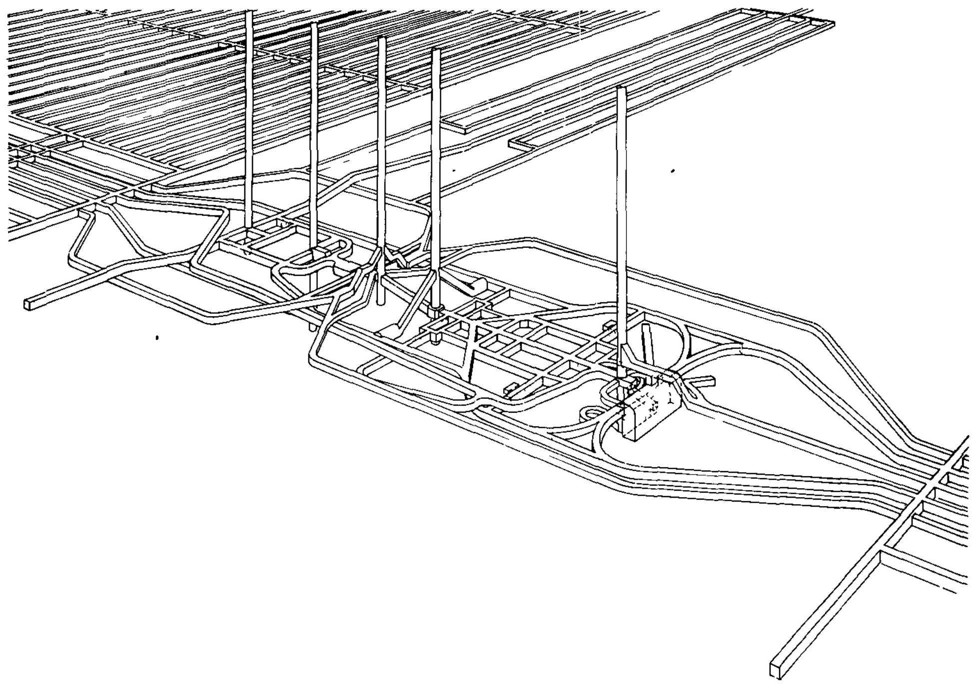

Figure 8 Shaft pillar.

rock, and any water that may seep in. Storage rooms will not be backfilled until after the retrievable period. The behavior of the backfill will be tested in an experimental panel for as long a time as may be required.

Waste packages are handled in the repository in transfer casks that reduce radiation levels to 10 mrem/hr at 0.9 m from the surface. A transfer cask is shown in Fig. 17.

Waste packages are emplaced by a diesel-powered, rubber-tired transporter (Fig. 18), which receives a transfer cask at the base of the waste transport shaft, carries it to the storage location in a horizontal position to minimize excavation requirements, and then raises it to vertical. After the transporter sets a floor shield in place over the storage hole, a shield door is opened at the bottom of the transfer cask and the waste package is lowered into storage.

Surface Facilities

A plan of the surface facilities, where waste is received at the repository (Fig. 19), and an artist's conception (Fig. 20) show the extent of the basalt storage pile (472 by 472 by 46 m), which will contain approximately 13.6 million tonnes before the start of backfilling. Figure 21 gives a closer view of the buildings in the central process area and identifies their functions. The most prominent building is the waste handling building, a cutaway view of which is shown in Fig. 22.

Figure 23 shows details of the canister receiving and handling areas in the waste handling building. It shows a canister being unloaded from a shipping cask on a rail car. The shipping cask was horizontal during transit and has been raised to a vertical position for unloading and then mated with a

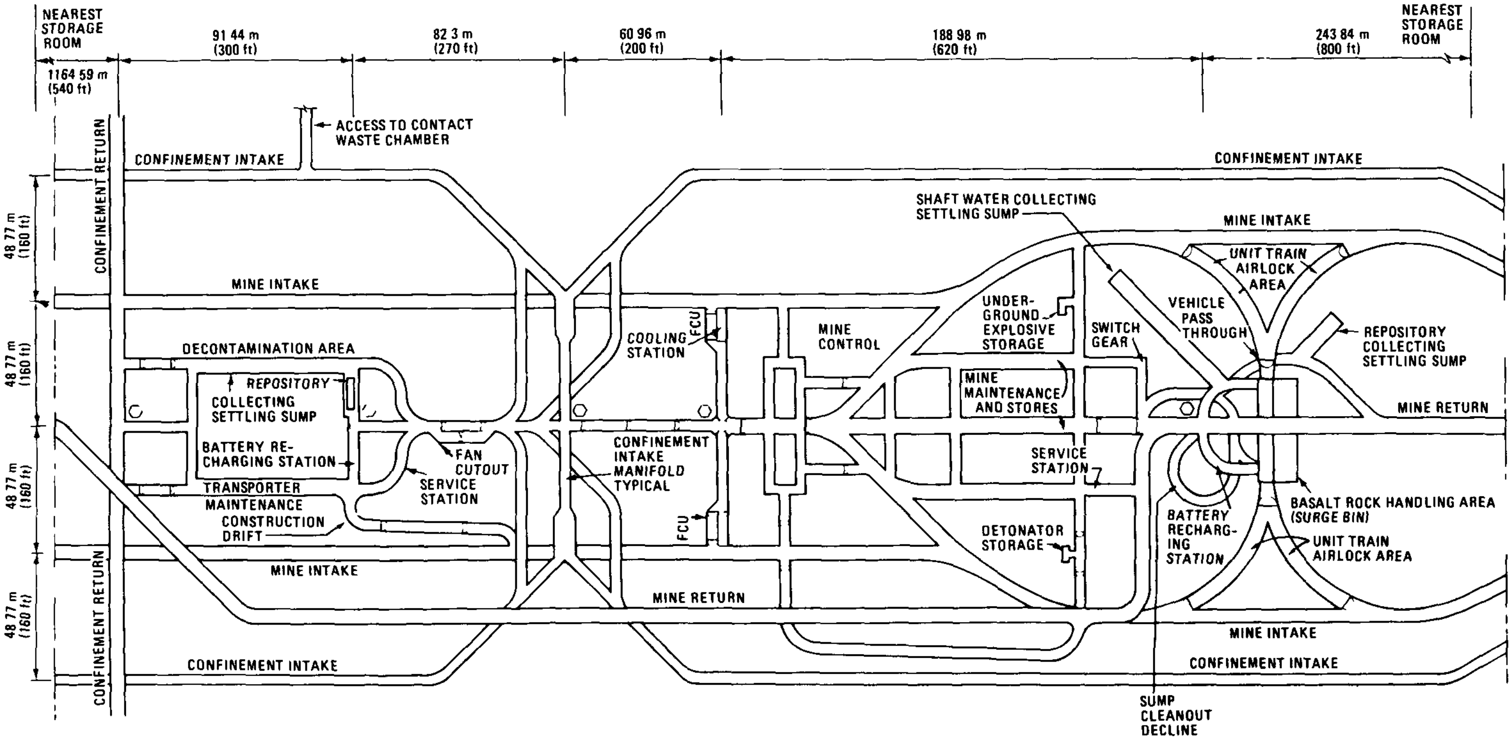

Figure 9 Shaft pillar layout. 8, fan; D, airlock door.

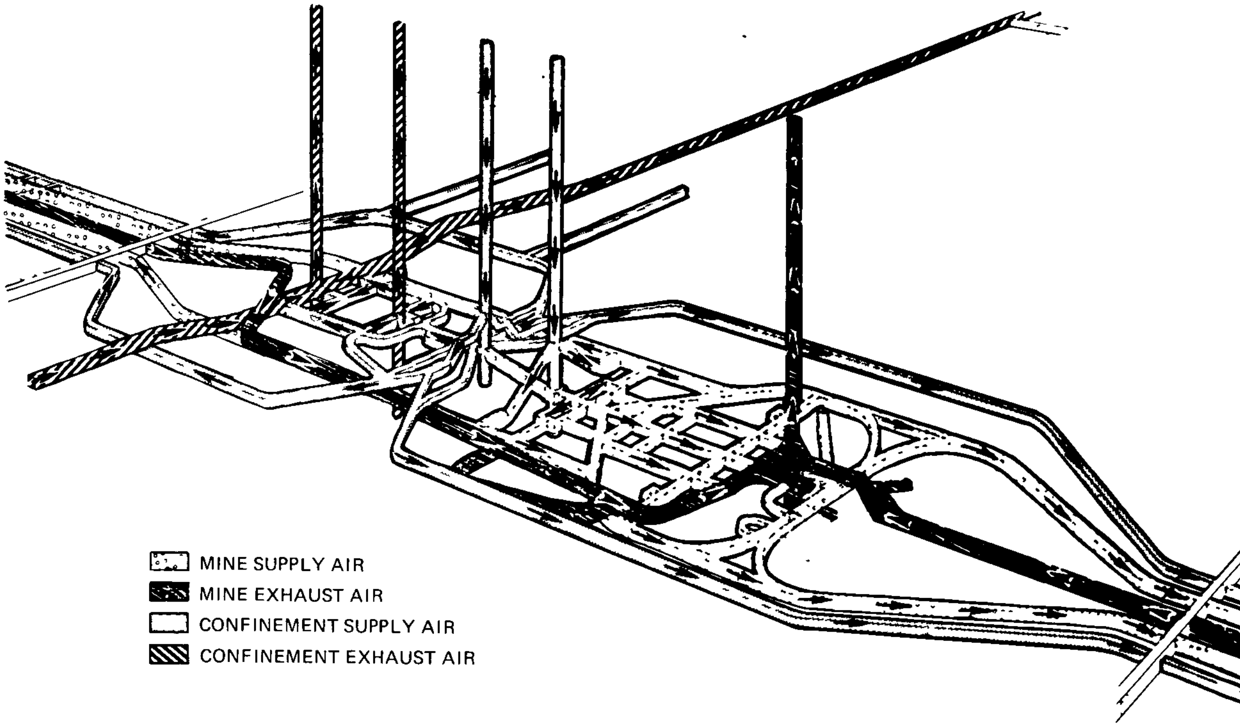

Figure 10 Shaft pillar flow paths.

Table 3 Subsurface Ventilation Requirements

Shafts					
Designation	Description	Size, m	Air quantity, m^3/min	Air velocity, m/min Criteria	Air velocity, m/min Actual
1	Confinement exhaust shaft	3.05	6460	914	884
2	Waste handling shaft	3.66	708		
3	Confinement intake shaft	3.35	6000	762	759
4	Service shaft	4.88	7220	610	436
5	Basalt hoisting shaft	4.27	6970	610	549

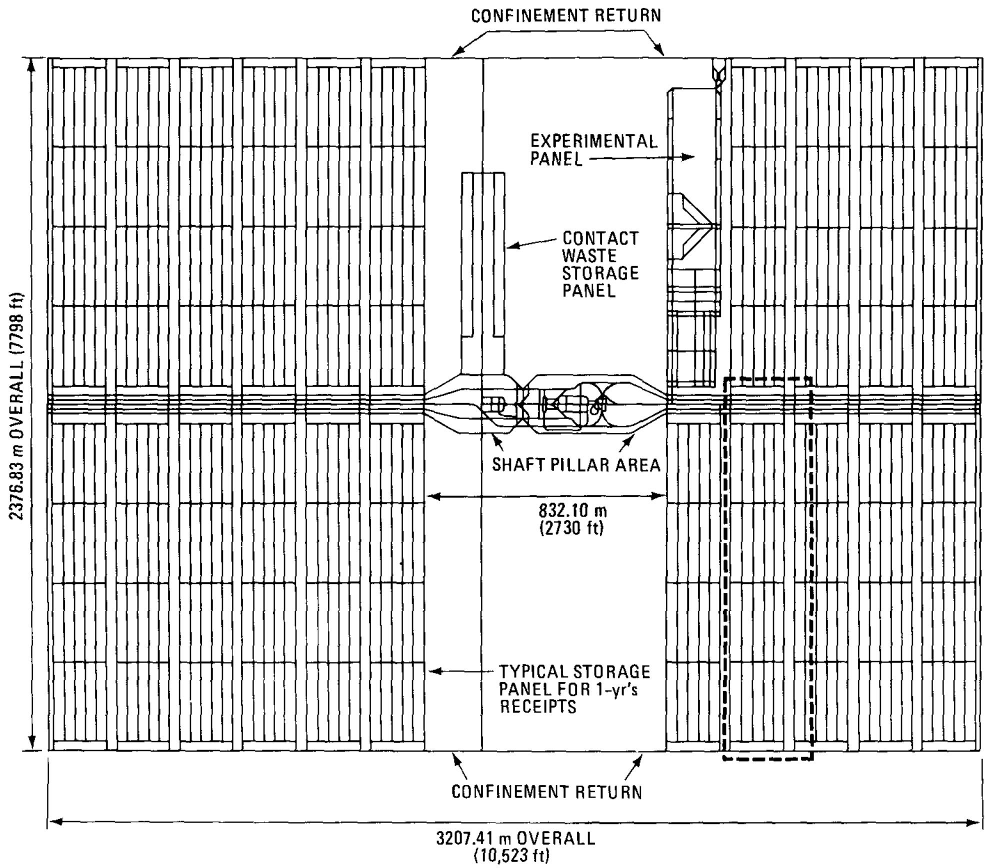

Figure 11 Layout of underground facilities.

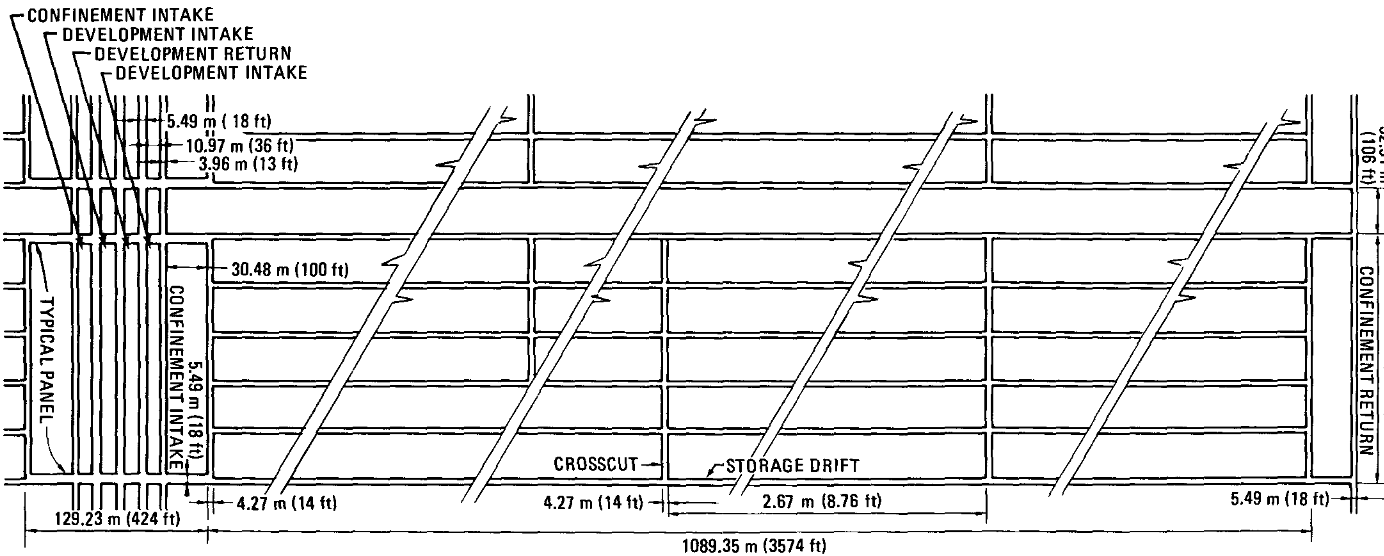

Figure 12 Typical storage panel.

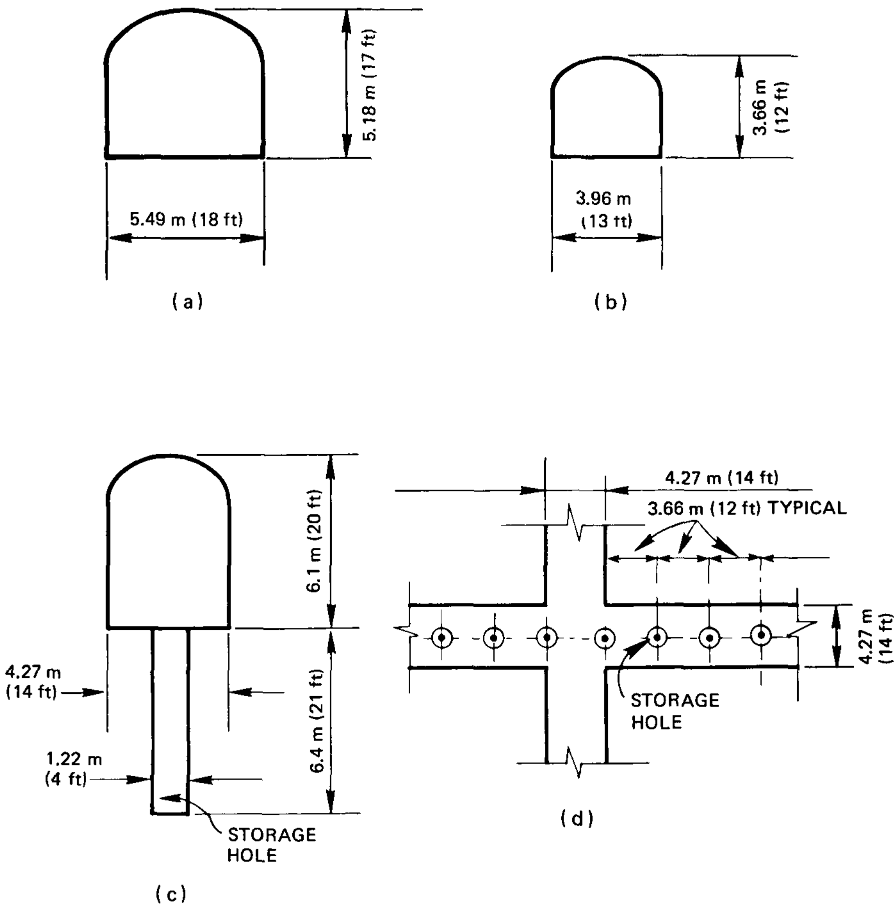

Figure 13 Cross sections and plan of rooms. (a) Mine entry section. (b) Confinement entry section. (c) Storage room section. (d) Storage room plan.

shielding collar lowered from the hot cell. A canister mover in the hot cell is shown removing the canister from the shipping cask. In the floor to the left of the canister mover are 20 surge storage positions for canisters or waste packages. Slightly farther to the left in the floor is a process tank in which the canister is welded into the overpack to form a waste package. Completed waste packages are positioned below a port at the left end of the hot cell; a transfer cask on a mover above the hot cell then picks up the waste package and moves it further left for placement in the cage in the waste transport shaft.

Thermomechanical Considerations

Thermomechanical analyses are required to:

- Ascertain safety and rock support requirements for excavation
- Make sure temperatures do not exceed prescribed limits

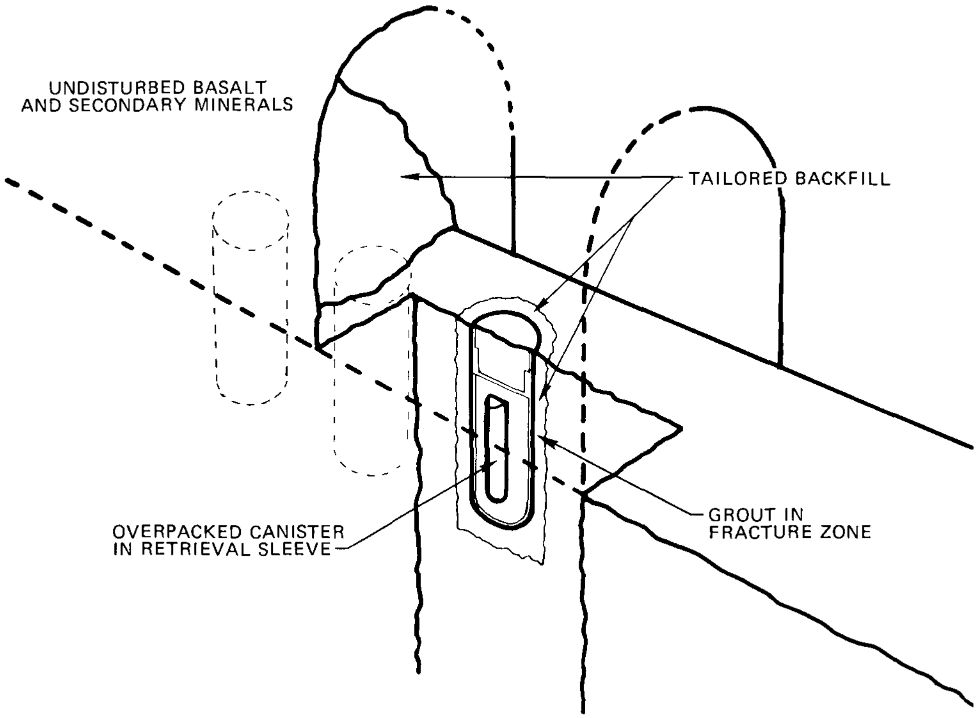

Figure 14 Storage room.

- Evaluate requirements for and effects of ventilation cooling
- Predict stresses and strains around openings that may alter permeabilities
- Assess the risk that tensile stresses at a distance may open paths for water ingress or egress
- Assess the risk of rock bursts or induced seismicity that may alter permeabilities
- Determine probable or potential paths for escape of radionuclides and probable radionuclide travel times, considering also absorption and adsorption

Figure 24 shows the thermal basis for the spacing of the spent fuel. The initial basalt temperature is 57°C, and the maximum allowable temperature for basalt under the criteria is 500°C. The maximum permissible fuel cladding temperature is 300°C. The temperature drop from the fuel rod nearest the canister center line to the basalt at the edge of the storage hole is 100°C. Thus the cladding temperature criterion governs, and the basalt temperature should not exceed 200°C.

Several different alternative vertical and horizontal storage arrangements for the waste packages were analyzed. As a result of these analyses and considering the technical and economic advantages of minimum removal of the host rock, the storage configuration selected placed the waste packages in the floor of a room 4.3 m wide by 6.1 m high with a 3.7-m spacing (pitch) between waste package center lines and a 36.6-m separation between rows. The results of the thermal analyses used in the selection process are shown in Fig. 25. The transient rock temperatures at locations adjacent to the waste package and

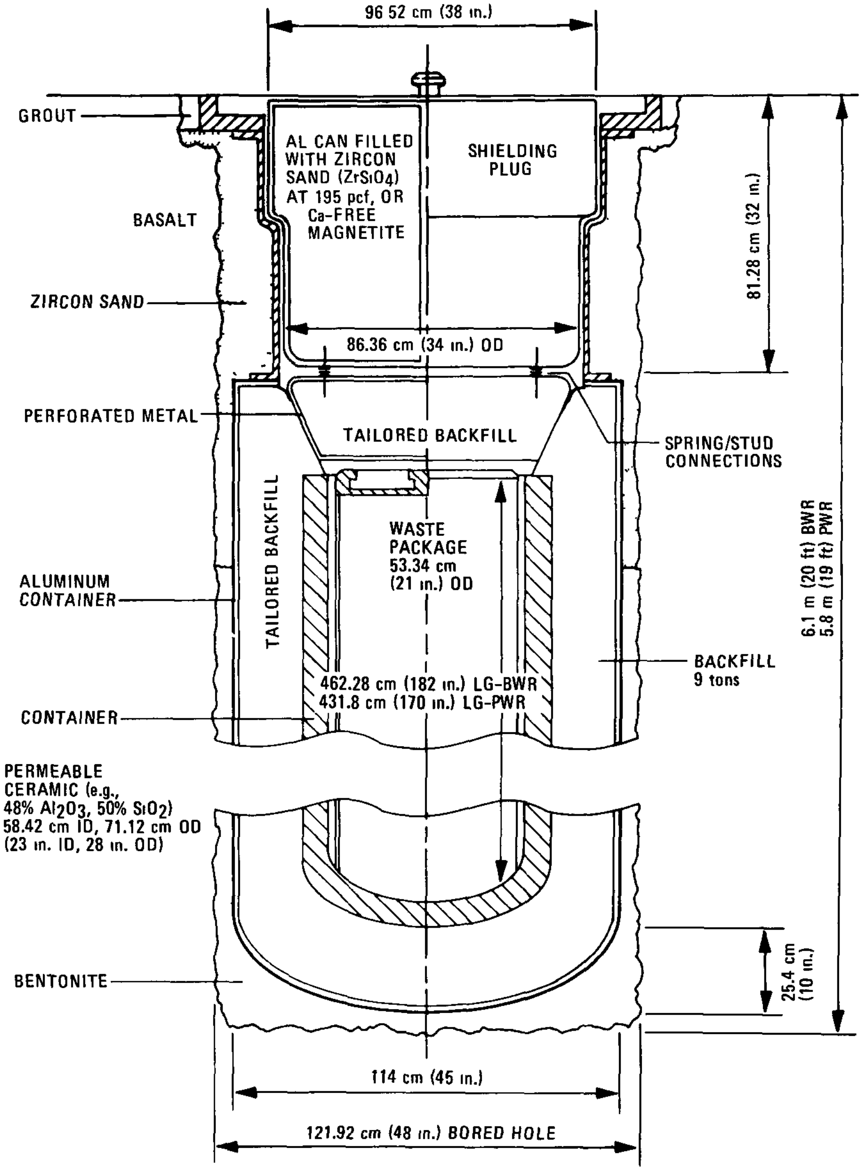

Figure 15 Storage position in Nuclear Waste Repository in Basalt.

around the storage room for the selected configuration are shown in Fig. 26. The rock temperature profiles at various elevations across the pillar that will separate the storage rooms after 25 yr of storage is shown in Fig. 27. As shown in Fig. 26, the major portion of the temperature increase in the near field occurs during the first 25 yr of storage.

Because of the relatively high initial basalt temperature and the heat

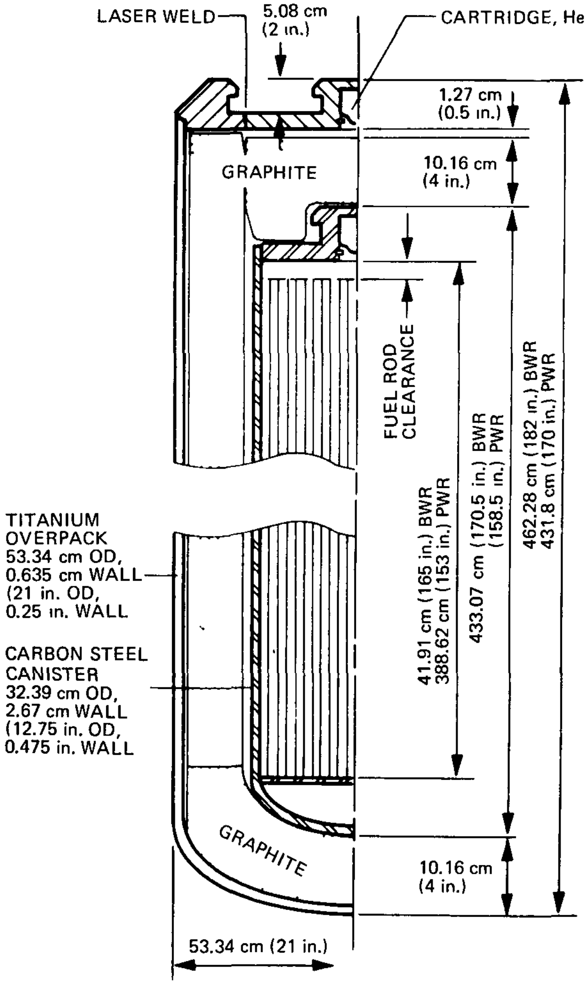

Figure 16 Waste package.

added by the spent fuel, cooling of the repository is required for habitability, both in the development and confinement areas. A simplified flow diagram for the cooling water system is shown in Fig. 28. Chilled water supply and return lines in the confinement intake shaft (Fig. 5) circulate water through heat exchangers at the storage level. The heat exchangers isolate the secondary chilled water system in the repository from the 6.23-MPa hydrostatic pressure at the bottom of the primary system. The total repository cooling load is 3400 tons; however, **200 tons of heat removal is provided by service water to drilling and mucking operatons. Thus the capacity of the refrigeration plant is 3200 tons.**

Thermal rock properties in small sample sizes are known or may be determined quite accurately (±10 to 20%); they are believed to be only moderately affected by rock flaws or jointing. If the water content of the rock is small and the rock has a low permeability, heat transfer calcula-

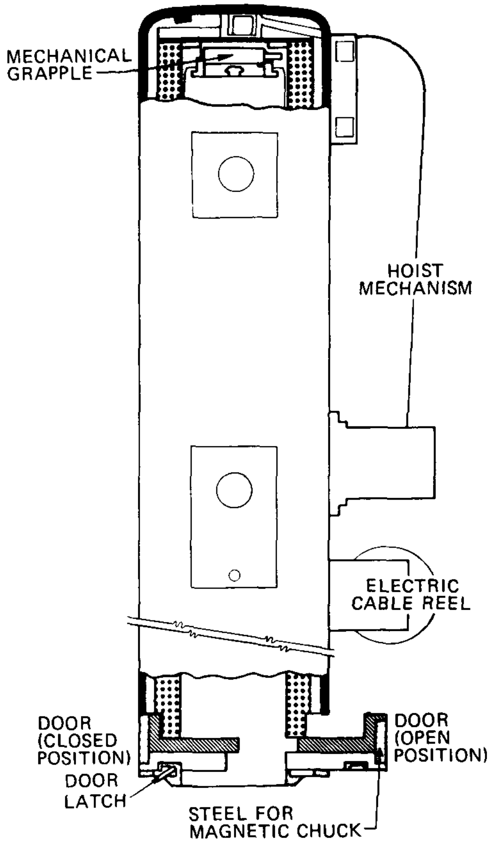

Figure 17 Transfer cask.

tions are relatively simple and reliable, and temperatures can be predicted with confidence. As long as the rooms are open, air convection in them will tend to equalize temperatures immediately around the rooms and evaporating water will remove heat. These effects are difficult to evaluate but are likely to render rooms safer; they may, therefore, be ignored for conceptual design analyses.

Rock Stress Safety Factors

Openings in rock do not behave like structural members, for which a failure load can be determined. For a rock opening, the load comes from the rock itself, which also is the principal supporting member. No failure load can be defined, and a safety factor similar to that of an ordinary structure cannot be calculated. Instead of safety factors, we must deal with other expressions of satisfactory or unsatisfactory performance defined to suit circumstances and use requirements.

For rooms in which workers are present, safety requires a low probability of one or several modes of failure, such as rock falls, wall slabbing, or rock bursts. During the period of retrievability, during which rooms are sealed and are not routinely accessible, performance is satisfactory if no

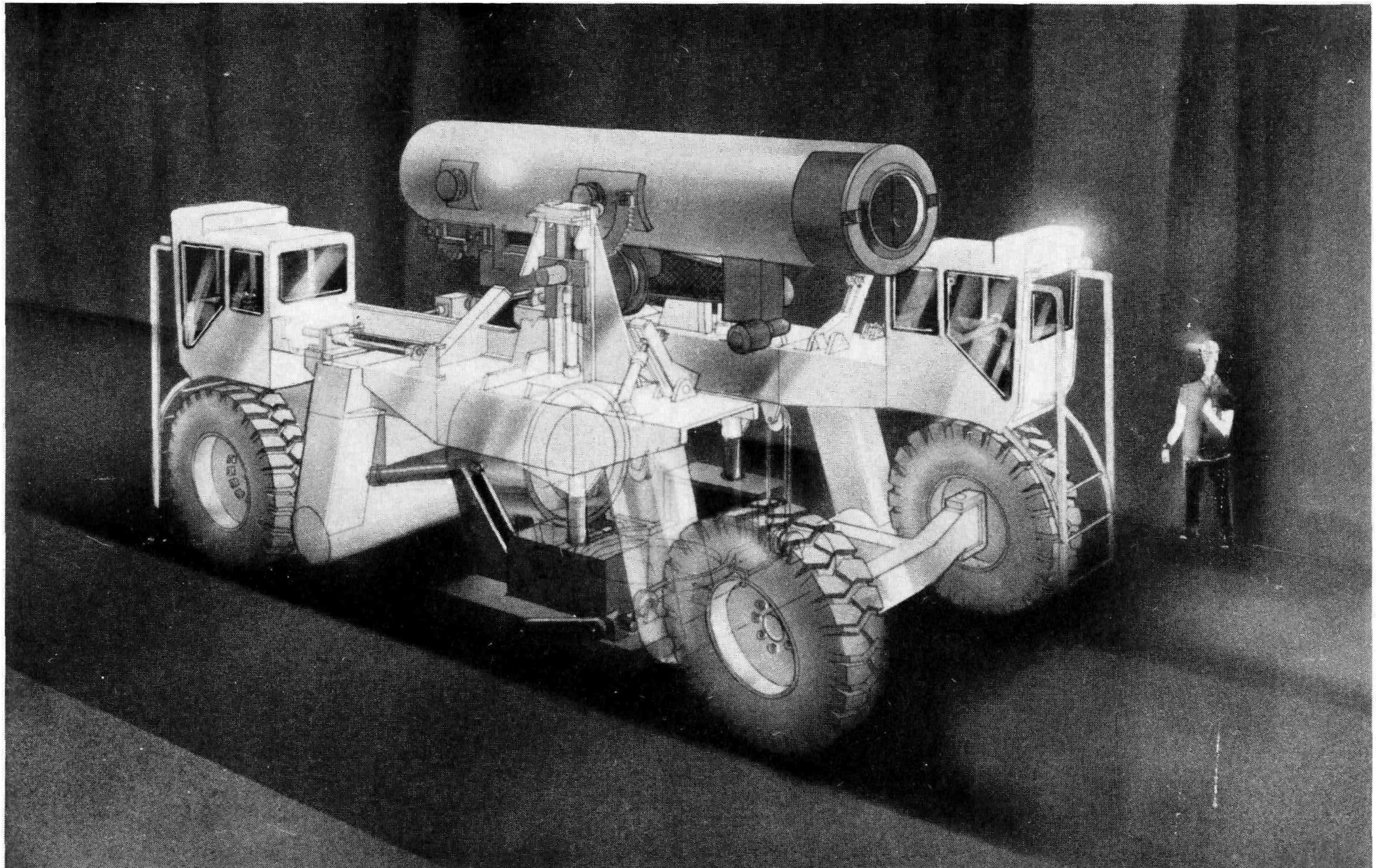

Figure 18 Waste transporter.

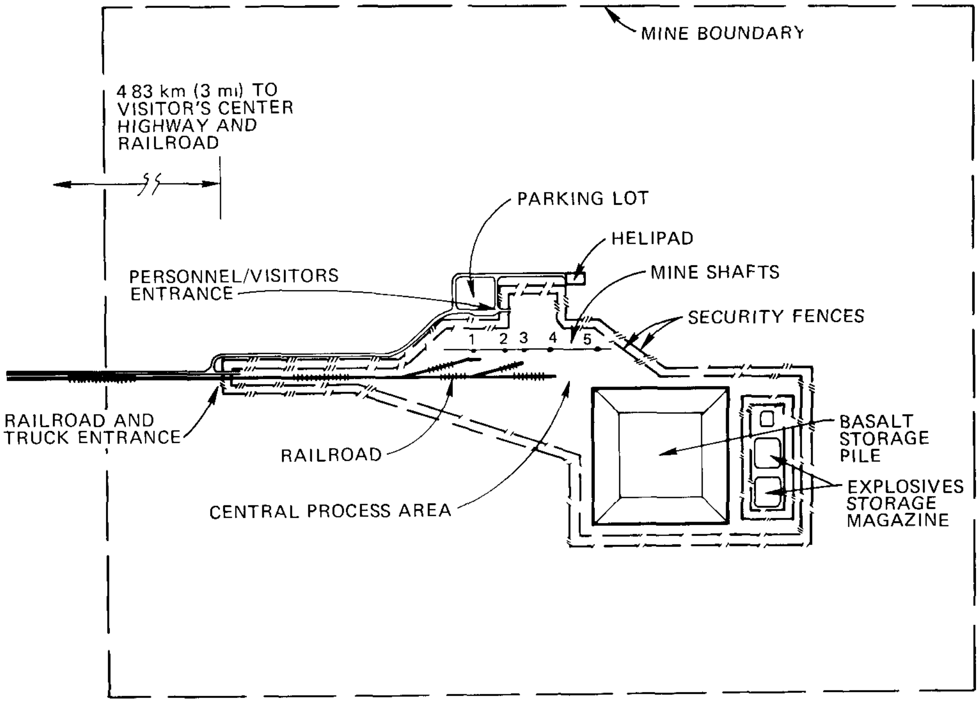

Figure 19 Overall plan of surface facilities.

major collapses occur and if instability can be checked and repaired safely upon reentry to the rooms. In the long term, safety against radionuclide migration requires that no continuous paths of high permeability be created that might quickly carry groundwater containing radionuclides to the biosphere or that such permeable paths are checked by a barrier at some point.

Room safety assessments begin by determining acceptable modes of performance. These may relate to purely mechanical phenomena (e.g., initial rock fall probability or thermomechanical phenomena, i.e., mechanical stability under stresses induced by higher temperatures), to changes in hydraulic characteristics induced by thermomechanical phenomena, or to hydrothermal phenomena (chemical modifications of rocks, joint infills, or man-made materials). Borehole decrepitation (e.g., by thermal spalling) is not ordinarily a problem in basalt, which is essentially nonspallable (Thirumalai, 1970), but may be a problem in other rock types.

Next, analyses must be performed which specifically address the phenomenon under investigation, or experience must be studied to determine the probability of unsatisfactory performance. Most such analyses performed today are deterministic in nature. Based on realistic input, the analyses produce predictions of performance that are either satisfactory or unsatisfactory. It is likely that probabilistic analyses, which are based on probabilistic data input, will produce an estimate of the likelihood of satisfactory performance or of the proportion of room length for which unsatis-

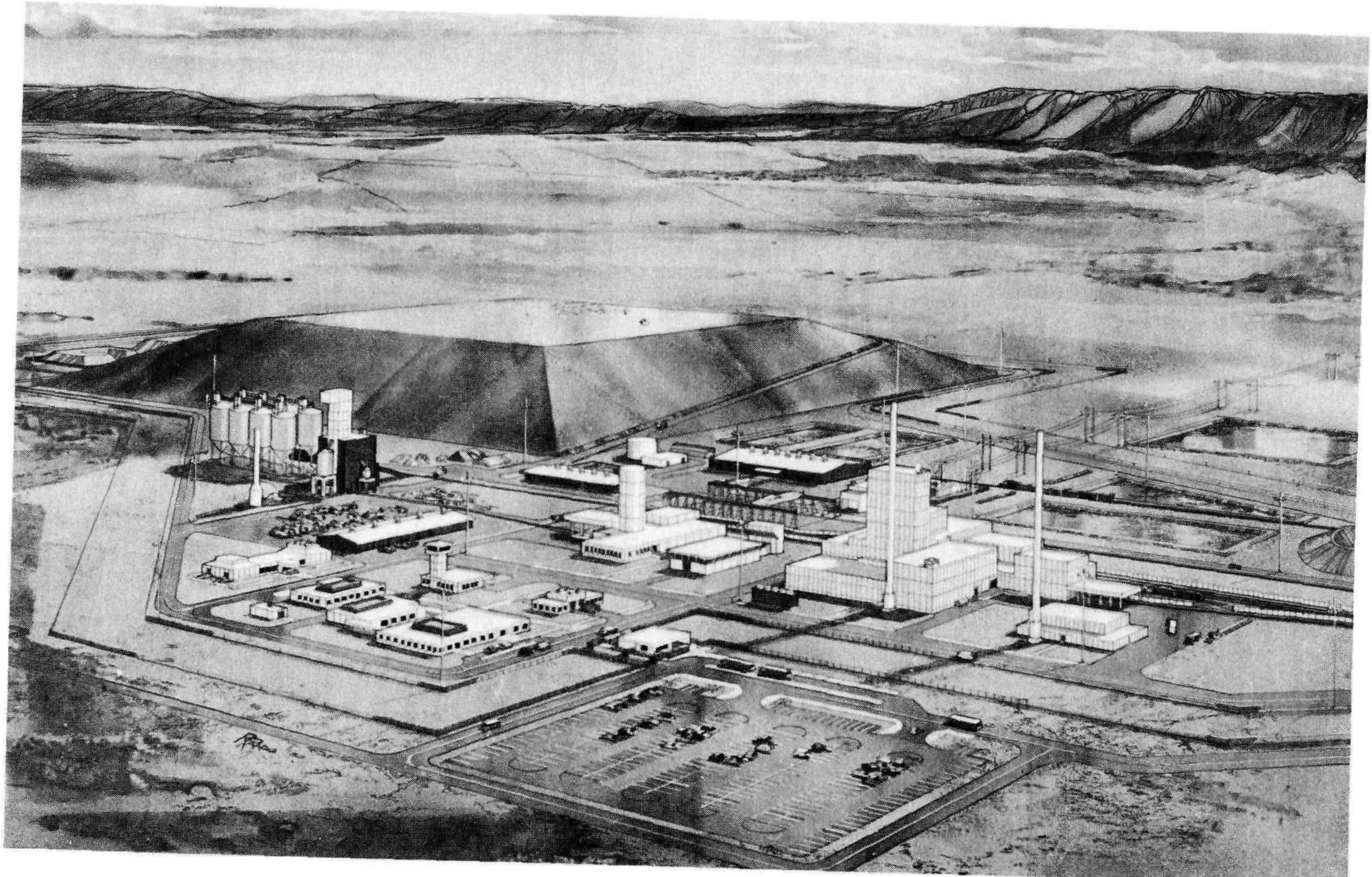

Figure 20 Artist's conception of surface facilities.

1. PERSONNEL/VISITORS ROAD
2. TRUCK ROAD
3. SECURITY FENCES
4. PARKING LOT
5. GATE HOUSE NO. 1
6. HELIPAD
7. ADMINISTRATION BUILDING
8. CAFETERIA
9. SUBSTATION
10. INDUSTRIAL SAFETY
11. TRAINING CENTER
12. SECURITY HEADQUARTERS
13. FIRE STATION
14. CONFINEMENT EXHAUST BUILDING
15. STACK
16. SHAFT NO. 1
17. COOLING TOWERS
18. STACK
19. SHAFT NO. 2
20. WASTE HANDLING BUILDING
21. CASK CAR STORAGE YARD
22. REFRIGERATION BUILDING
23. SHAFT NO. 3
24. CONFINEMENT AIR INTAKE BUILDING
25. PERSONNEL AND MATERIAL ACCESS FACILITY
26. SHAFT NO. 4
27. MINE AIR INTAKE
28. MAINTENANCE
29. SHAFT NO. 5
30. STACK
31. MINE EXHAUST BUILDING
32. BASALT AND MATERIALS HANDLING BUILDING
33. PATROL ROAD
34. BASALT CRUSHER BUILDING
35. CORE STORAGE AND LABORATORY BUILDING
36. SLEEVE AND PLUG STORAGE YARD
37. SUSPECT RAIL CAR AND TRUCK STORAGE AREA
38. STANDBY GENERATOR BUILDING
39. MINE WATER RETENTION PONDS
40. RECEIVING SUBSTATION
41. WAREHOUSE
42. SANITARY SEWAGE TREATMENT PLANT
43. PROCESS WASTE EVAPORATION PONDS
44. OVERHEAD POWER LINES
45. MINE WATER PERCOLATION POND
46. BASALT HAUL POND
47. BASALT STORAGE PILE

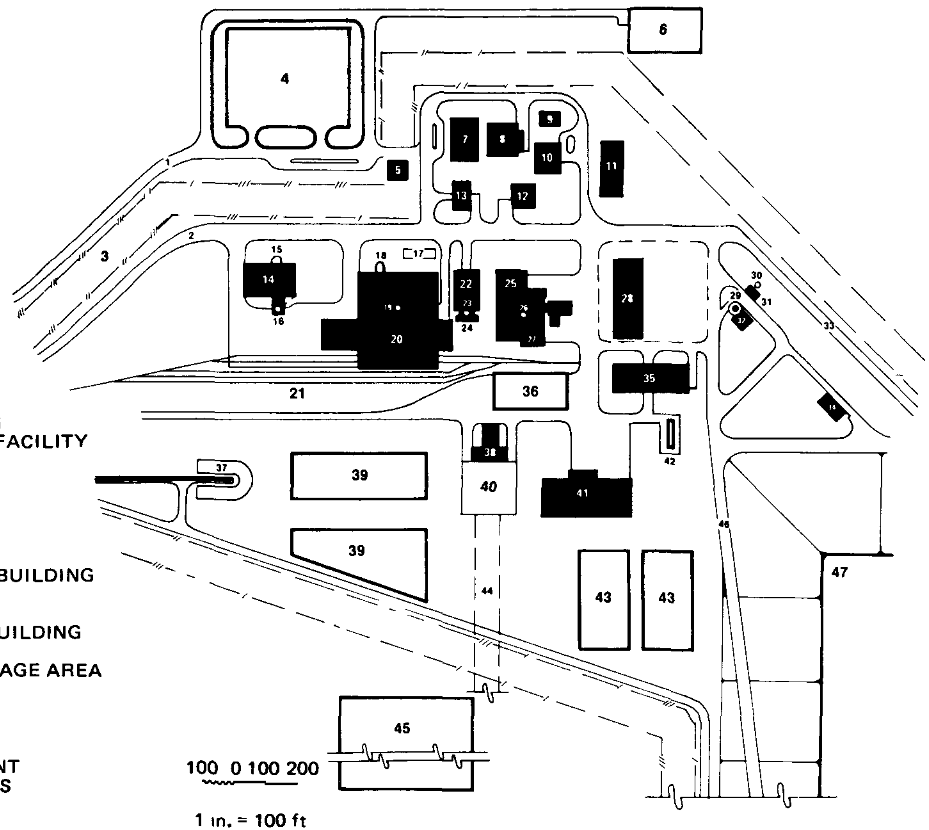

Figure 21 Plan of central processing area.

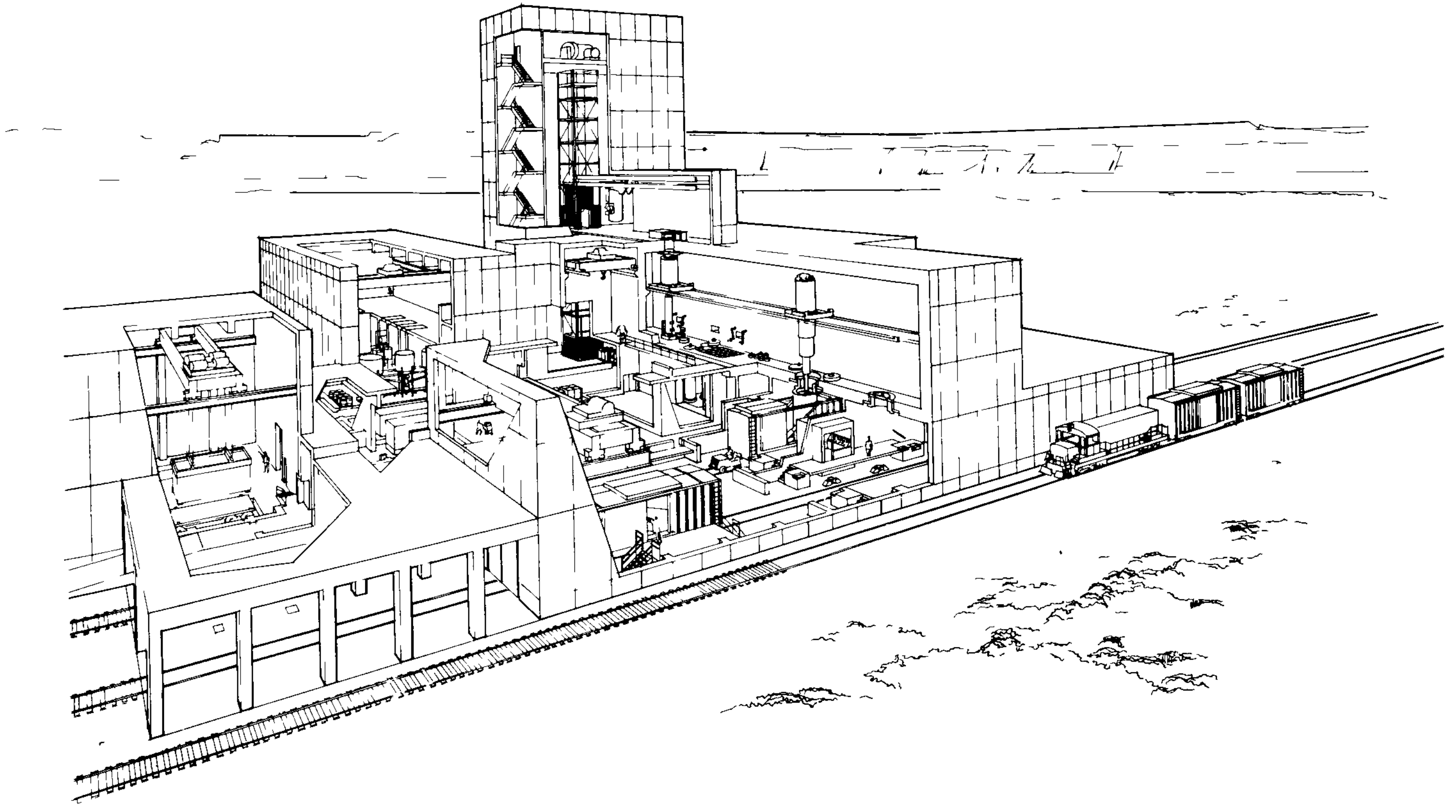

Figure 22 Waste handling building (cutaway view).

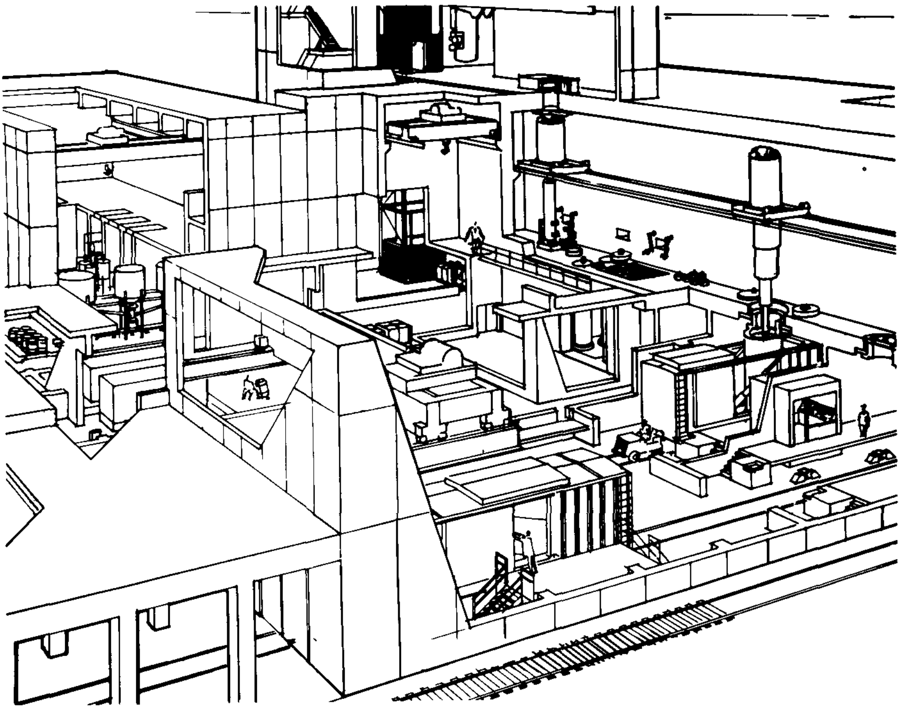

Figure 23 Canister receiving and handling.

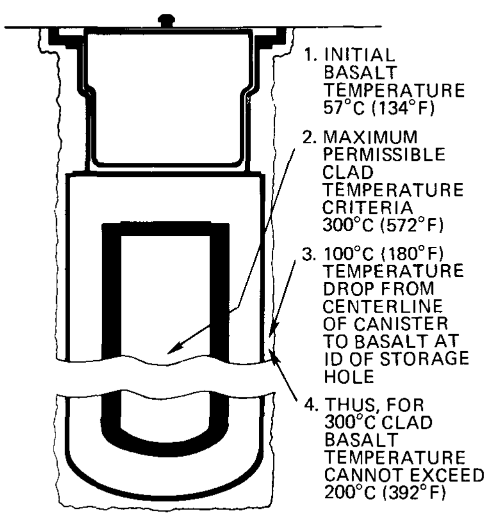

Figure 24 Heat transfer. Maximum permissible basalt temperature, 500°C. The temperature of basalt in this case, 200°C, sets the pitch at 3.66 m and the row separation at 36.6 m.

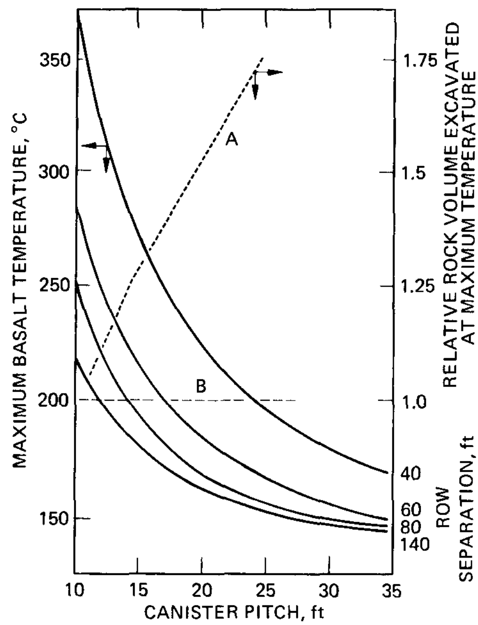

Figure 25 Vertical selection of pitch and row separation for a single row of waste canisters with three PWR fuel elements per canister. A, relative rock excavation for varying pitch and row separation yielding 200°C maximum temperatures. B, maximum allowable basalt temperature of 200°C to meet maximum allowable cladding temperature of 300°C.

factory performance may be expected. It must then be judged if the predicted probability is acceptable.

Such types of analyses are difficult to make today, because a number of parameters are as yet poorly defined and rock mass behavior in all its complexity is poorly understood. The strength of intact rock and its variability may be tested adequately in the laboratory, but that of the jointed rock mass is much less tangible. Progress is being made to characterize rock mass strength on an empirical basis (e.g., Hoek and Brown, 1980), but little is known about rock mass strength in probabilistic formulation. Stresses induced by temperature increases are proportional to Young's modulus; yet Young's modulus for rock masses is known with an accuracy of a factor of only three at best, and no probabilistic formulation is available.

For the moment, then, we must make do with relatively simple but imperfect deterministic analyses. Simplified hand calculations are adequate for scoping of conceptual designs. More detailed designs require the use of computer models.

Much of the analysis approach for the NWRB conceptual design follows principles described by Hardy and Hocking (1980). With this approach, a rock stress safety factor is estimated for a given point by calculating the stresses and estimating the strength, considering the confining stress, at the same point. The rock stress safety factor is simply the strength divided by the stress. A triaxial strength formulation is used, and a point located a distance of one-sixth of the roof span

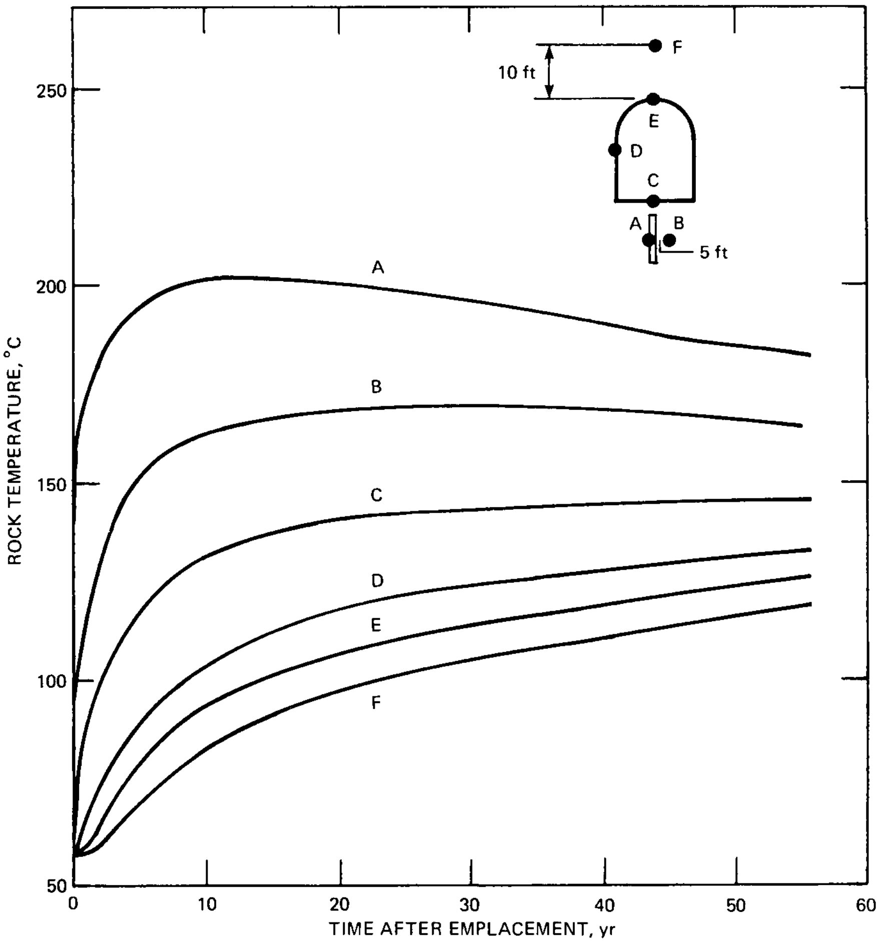

Figure 26 Rock temperature vs. time at various locations around the storage room.

away from the opening is considered. The rock strength is reduced as a function of representative volume or roof span. These principles are shown in Fig. 29. As noted, the rock stress safety factor is not a safety factor in the conventional sense; it is a relative measure of stress level.

Results of Thermal and Rock Stress Safety Factor Analyses

Twenty-five years after placement of 10-yr-old waste, with 1.75 kW of heat generated per waste package (131 kW/ha), the temperature distribution around a room would be approximately as shown in Fig. 30.

The temperature increase results in increased horizontal stresses since the rock is confined in a horizontal direction. By integrating the temperatures and applying the proper stress concentration factors, we can calculate stresses around the room and estimate the rock stress safety factor. The decrease in rock stress safety factor with time is shown in Fig. 31. The time variation of rock stress safety

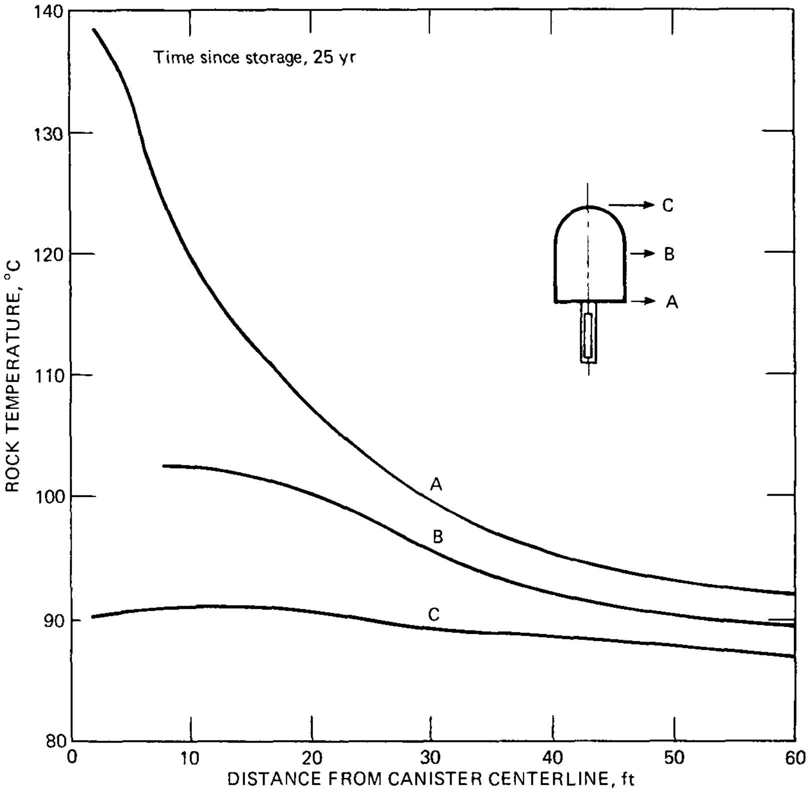

Figure 27 Pillar temperatures at various elevations.

factor for an equivalent circular room is also shown.

Initially the rock stress safety factor is about 2.7, but it decreases to about 2 after 50 yr. Beyond this time the decrease is small. The equivalent circular room is larger, and the representative strength, therefore, is smaller. The stress concentration is lower, however, because of a more favorable shape; hence a circular room is just slightly safer.

The rock stress safety factors are used primarily to compare different room sizes, shapes and spacings and to determine the relative effects of temperature increases. A safety factor of 2 appears acceptable but does not guarantee that nominal overstress would not occur locally. More detailed analyses will show whether unsatisfactory performance (if any) is likely to occur in an acceptably small proportion of the room lengths.

Some additional findings of these and similar analyses are discussed in the following subsections.

Rock Support. The effect of rock support on the theoretical rock stress safety factor for rooms at great depth is small. Maximum stress concentrations around the room periphery reach beyond 70 MPa. If room safety is not acceptable, even a modest improvement in the theoretical rock stress safety factor would require supporting loads of many megapascals. This would easily exceed the capacity of practical support materials. Hence, ideally, rooms should be designed so that the rock stress safety factor is

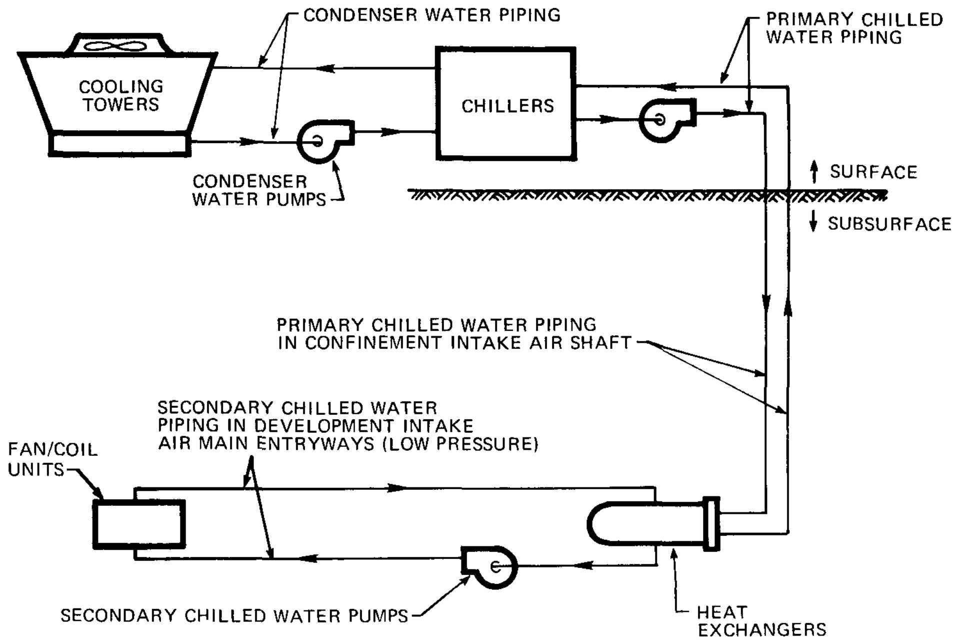

Figure 28 Cooling water system for repository.

adequate even without considering rock support. In practice, however, experience shows that, with proper application of supports, rooms can be made stable even in severely stressed rock.

Minor Modes of Failure. Even if the theoretical rock stress safety factor is acceptable, supports are likely to be required. Joints frequently occur in unfavorable locations and directions and lead to potential minor modes of failure, such as roof falls, unacceptable loosening, or wall slabbing. These minor failures can be prevented by a shotcrete application or rock bolts in a regular pattern or both. These supports would be designed according to experience gained elsewhere and during development of the repository.

After a room is excavated and shotcrete is applied, an initial safety factor better than unity is assured everywhere. Subsequent installation of rock bolts, designed essentially to duplicate the supporting effect of the shotcrete, will add desired conservatism to the design and preserve safety during the retrievability period, when roof stresses will increase. The order of application of shotcrete or rock bolts may be reversed without changing the effect of redundancy. Although these supports would be designed in essentially empirical fashion, modeling tools are available to assess the redundancy effect.

Effects of Temperature and Initial Stresses. Under the initial stress conditions at a depth of 1128 m, the safety of the rooms would decrease by about one-third with increasing temperature. This decrease is nearly proportional to the overall heat load supplied by the waste packages. To maintain satisfactory safety, therefore, an upper limit may be applied to the areal heat load. For NWRB the initial heat load is about 131 kW/ha (for waste packages with a pitch or spacing of 3.7 m in rooms

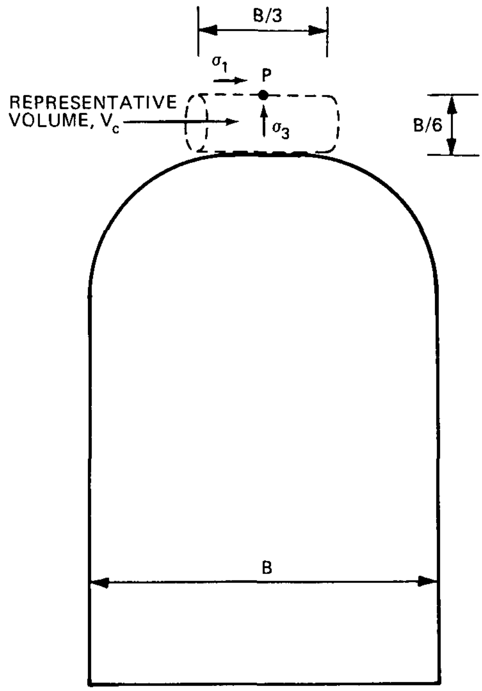

Figure 29 Rock stress safety factor, which is strength of representative volume (confined) divided by maximum principal stress at P.
Effect of volume on strength: $\sigma_c = \sigma_{Lab} (V_{Lab}/V_c)^{0.1}$
Effect of confinement on strength: $\sigma_1 = \sigma_c + A\sigma_3^{0.75}\, \sigma_c^{0.25}$
A = 3.0 for initial calculations.

spaced 36.6 m apart). Doubling the heat load reduces the long-term safety by two-thirds; this does not appear acceptable.

If the initial stresses were significantly lower (e.g., if the repository were at a shallower depth), the initial safety would be greater, and a greater reduction in safety with time could be accepted. Hence waste packages could be placed closer together in a shallower repository, and thus a shallower repository would be much less expensive.

The conceptual design has been worked out under the assumption of equal initial vertical and horizontal stresses. If additional investigations in an exploratory shaft test facility disclose significantly higher in situ horizontal stresses, the density of waste placement might have to be reduced. Changes in room shapes and layouts may also be desirable.

Effects of Stratification. The repository would be located within a basalt flow 60 to 90 m thick. As discussed earlier, the strength of the rock and its Young's modulus are lower by nearly an order of magnitude near the flow top and in the interflow region. This significantly reduces the capability of the rock to transfer shear strains from one basalt flow to the next above or below. Hence horizontal compressive stresses tend to stay within the host basalt flow and are not redistributed into adjacent flows. This would increase the hor-

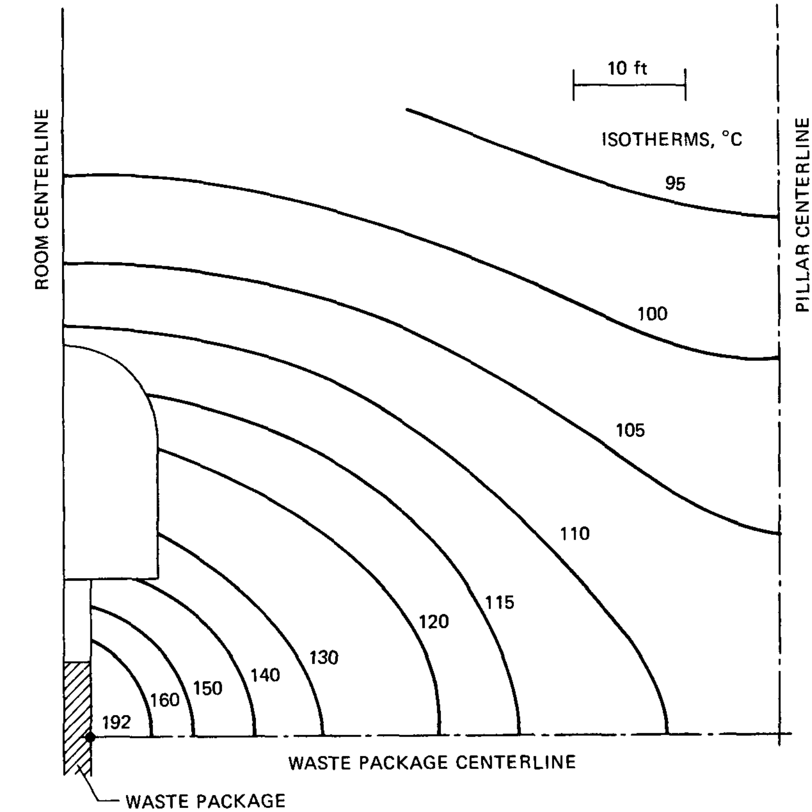

Figure 30 Temperatures after 25 yr. Initial temperature was 57°C.

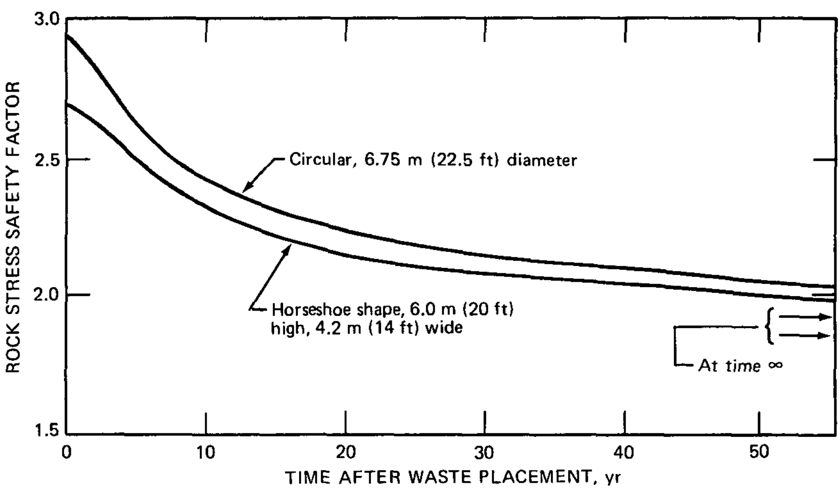

Figure 31 Variation of rock stress safety factor with time.

izontal thermal stresses within the flow beyond what they would be in a homogeneous mass. On the other hand, the likelihood of creating tensile cracks some distance above the repository is diminished (that could be deleterious to long-term sealing).

Maintenance During the Retrievability Period

The repository rooms can provide adequate safety throughout the retrievability period without additional support. Any backfill material is likely to have a strength and modulus an order of magnitude lower than that of the surrounding rock mass. Hence support from the backfill is likely to be only nominal. In retrieval the removal of backfill would be difficult and hazardous. The backfill would be quite hot (its temperature would be greater than 100°C) and, therefore, would require substantial ventilation and cooling efforts. Latent rock falls could be discovered and fixed if the rooms remain open. If rooms are backfilled, however, removing the backfill could expose workers to such hazardous rock falls without warning.

For these reasons it appears that rooms should not be backfilled (though they may be sealed) until after the retrievability period or until the probability that retrieval will be required is very small. The long-term performance of a backfilled room can be demonstrated in a dedicated experimental panel; backfilling of all rooms immediately after waste placement is not required for this purpose.

Monitoring

Two different philosophies can be applied to the subject of monitoring repository behavior after waste placement:

- Monitor to ascertain satisfactory performance of every point or cross section of every room and every waste package.
- Monitor to obtain confidence that predictions of safety are acceptably accurate.

The first philosophy places no reliance on the ability to predict behavior or to assess the gross effects of locally unacceptable performance; it ignores concepts of probability. Monitoring to satisfy this philosophy for a repository containing over 160 km of rooms and 35,000 waste packages for tens of years would be extraordinarily expensive and essentially infeasible.

If we use a probabilistic approach and consider that locally unsatisfactory performance may be quite acceptable for the repository as a whole, the second approach is favored. The proposed program uses a full complement of monitoring devices in a dedicated experimental panel and in a limited portion of the production panels. For the remainder of the panels, monitoring would be used only to provide adequate statistical testing of panel behavior.

The monitoring program is still under development. It will include monitoring of microseisms, temperatures, room convergence, rock strains, and displacements by extensometers and rock stress gauges. Water inflow into rooms will be monitored by measuring liquid and vaporized outflow from rooms. Small volumes of air pulled through repository rooms will be monitored for radionuclide contents, as will the drainage water (if any).

Though not every point of the repository would be monitored directly, areal monitoring systems based on acoustic emissions would detect adverse rock movements at any location. Radionuclide leaks would be detected by monitoring air and water from each room or panel. Further investigation of suspected areas would then lead to discovery and remedial action.

Rock Disturbance Caused by Construction

The mechanisms of rock disturbance around a room excavated in rock are complex and poorly understood. For another type of project, this may be of minor consequence since construction experience provides the means for designing supports for adequate safety. For a nuclear waste repository, the potential for increased permeability in a disturbed zone has significant effects on the design of schemes for eventual plugging. Such a permeable zone might create a water flow bypass around any plug placed within the opening.

Fractures created, extended, or opened by blasting may reach a distance of several feet, depending on rock conditions and blasting methods. The distance would be only moderately affected by the size of the opening. If necessary, special innovative excavation methods can be used to minimize this type of disturbance.

Irrespective of excavation methods, the creation of a rock opening causes stress changes and rock strains that may lead to minute motions along existing or new joints. Laboratory experiments have shown that microcracking in an intact, compressive test sample greatly increases when shear stress exceeds about 50% of failure stress. A similar phenomenon can be expected for a jointed rock mass. Hence concern for this type of rock disturbance may require the imposition of maximum stress levels around openings or of a rock stress safety factor. The required minimum rock stress safety factor cannot be stated with confidence at this time because not enough is known of the effects of such disturbances on rock permeability. Very likely, in situ measurements in the host horizon would be required to resolve this problem.

Finally, increasing stresses as a result of thermal loading could cause additional rock damage as the rock stress safety factor decreases. This would happen primarily in the emplacement rooms and only to a minor degree in access ways and shafts. Since the most important plugs would be placed in the access ways and shafts, this damage is of lesser concern than the initial construction damage.

When these phenomena are better understood after in situ testing, proper rock stress safety factors can be established. Safety factors would be different for initial conditions and later thermal conditions and would vary with location within the repository.

Conclusions

Deep basalt strata at Hanford present a suitable geologic medium for nuclear waste disposal. They are strong, practically impermeable, and remote from rapidly moving groundwater and usable aquifers.

Acceptable rock stress safety factors can be determined by disturbance and permeability criteria rather than rock stability. Such factors and criteria must be verified by in situ experiments.

Repository performance should be monitored after waste placement to obtain the necessary confidence on a statistical basis, but not every point and waste package needs to be monitored. Backfilling rooms in basalt is not desirable until the likelihood of required retrieval has become very small.

Stratification of basalt rock tends to concentrate thermal stresses within a single basalt flow. It appears that adequate safety for a repository can be attained in the candidate repository horizon at about 1128 m.

The next step is to investigate the actual conditions at depth by sinking an exploratory shaft and conducting a test and exploration program. Meanwhile additional work needs to be done in developing analytical tools to assess rock behavior on a statistical basis. Some parameters are poorly known at this time and must be assessed and factored into analyses and assessments before or during final design. These include:

- In situ stress state
- Effects of construction on rock disturbance
- Effects of rock stress concentrations and redistributions on rock disturbance
- Effects of rock disturbance on local permeability
- Applicability and calibration of models (including rock mass modulus calibrations)

Most of this information must be obtained in situ at the stress conditions prevailing at repository depth.

Some issues regarding the interface between the repository and the remainder of the total waste management system must be settled also. The approach taken during the NWRB conceptual design has been to make certain assumptions regarding these interfaces and, if necessary, to develop conceptual designs for them. Major issues that involve interfaces with the remainder of the waste management system external to the repository are:

- Design of the waste package and engineered barrier system. For the conceptual design, the functional design criteria specified a certain waste package configuration for a basalt repository. The configuration was developed further during the conceptual design, and the handling system and emplacement configuration was designed to accommodate this concept. A final waste-package-engineered-barrier system must be designed to match the particular requirements of a basalt repository, and the repository must be designed to handle the final waste package and other engineered barrier components.

- Waste form. The waste form assumed for the NWRB conceptual design is rods from disassembled PWR and BWR spent fuel assemblies. A change in waste form to, for instance, glassified high-level commercial waste would have a significant effect on the design of the repository.

- Transportation mode. The conceptual design assumed that high-level waste (spent fuel) would be received by rail or truck (predominantly rail). The final decision on transportation mode should be settled before initiation of final design.

- Packaging plant location. It was assumed, for the purposes of the conceptual design, that the packaging plant would not be colocated with the repository. The surface handling facilities in the current NWRB conceptual design have the capability to receive the canisters and overpack each canister of spent fuel before it is lowered to the storage horizon. Colocating a packaging plant with the respository would have a significant impact on the waste receiving and handling facilities in that certain features of the current NWRB conceptual design would not be required.

- Shipping cask design. In the NWRB conceptual design, a shipping cask was to be mounted on a rail car to transport the canisters of spent fuel from the packaging facility to the respository. Plans for this interface must be finalized before initiation of the final design for the basalt repository.

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Subseabed Radionuclide Migration Studies and Preliminary Repository Design Concepts

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Geochemical research carried out by the U. S. Subseabed Disposal Program is described. Data from studies of high-temperature interactions between sediments and pore water (seawater) and from studies of sorption and diffusion of radionuclides in oxidized, deep-sea sediments are used, along with results from heat transfer studies, to predict migration rates of radionuclides in a subseabed repository.

Preliminary results for most radionuclides in oxidized sediments are very encouraging. Fission products with moderate K_D values (10^2 to 10^5 ml/g) and actinides with high K_D values (10^3 to 10^6 ml/g) would not migrate significant distances before decaying to innocuous concentrations. Among this group are ^{137}Cs, ^{90}Sr, and ^{239}Pu. The results for anionic species in oxidized sediments are less encouraging. Planning for field verification of these laboratory and modeling studies is currently under way.

Conceptual repository designs and emplacement options are also described.

Introduction

The U. S. Subseabed Disposal Program (SDP) was begun in 1974 to assess the technical, environmental, and institutional feasibility of disposing of high-level waste or repackaged spent fuel in deep-sea sediments. One major objective of the SDP is to evaluate subseabed disposal as the primary alternative to mined geologic disposal of high-level waste, which is the option currently favored by the U. S. waste management community (Department of Energy, 1979, 1980a, 1980b). The choice of subseabed geologic formations as the primary alternative is justified by the fact that oceans cover roughly 70% of the earth's surface; these vast areas cannot be overlooked in a thorough search for solutions to the problems of radioactive waste management. Moreover, many deep-sea sedimentary formations reside in what is probably the most stable tectonic environment on earth, the midplate regions of the oceanic crust.

Recently the subseabed option has come to be viewed as complementary to mined geologic disposal. The removal of radioactive iodine from high-level waste and its disposal in subseabed sediments is an example of how the subseabed option might complement mined disposal (see Burger, 1980).

The other major objective of the SDP is to monitor the ocean-disposal activities of other nations. Even if the United States does not use the subseabed option, other nations, especially those of limited size or with unfavorable geologic environments, might resort to ocean disposal of high-level waste. Several nations are currently disposing of low-level waste at sea. This monitoring function is facilitated by U. S. participation in the Seabed Working Group, under the auspices of the Nuclear Energy Agency of the Organization for Economic Cooperation and Development. Other participants in the Seabed Working Group are shown in Table 1.

The U. S. SDP is divided into four phases:

- Phase 1, estimation of technical and environmental feasibility on the basis of historical data; completed in 1976
- Phase 2, determination of technical and environmental feasibility from newly acquired oceanographic and effects data; estimated completion date, 1988
- Phase 3, determination of engineering feasibility and legal and political acceptability; estimated completion date, 1993-1995
- Phase 4, demonstration of disposal facilities; estimated completion date, 2000

At the end of each phase of the program, a review, both internal and external, will be conducted to assess the feasibility of the concept. If any part of the concept is determined to be unacceptable, the program will be terminated, and efforts thereafter will concentrate on demonstrating this unacceptability to other nations considering the subseabed option for high-level waste disposal.

A three-part strategy is being used to assess the feasibility of the subseabed disposal concept: (1) development of mathematical models to predict effects over time periods that are often quite long (up to 10^6 yr); (2) acquisition of material properties data for use in predictive models; and (3) field verification of predictions made with the models and data. This strategy is illustrated in Fig. 1. Eventually models that predict various aspects of the response of a subseabed repository to high-level waste will be integrated and used to predict quantitatively the dose to man and the environmental effects that would result from the use of the subseabed option.

Most of this paper is devoted to a discussion of the radionuclide migration studies carried out in support of the SDP, which illustrate the three-part strategy. The remainder discusses repository design concepts and canister emplacement options. Other aspects of the SDP (site studies, thermal and mechanical properties of sediments, biological and physical oceanography, transportation studies, and national and international legal, political, and social issues) are described in detail by the SDP principal investigators (Bishop, 1975; Talbert, 1976, 1977, 1979a, 1979b, 1981b; Hinga, 1981).

Radionuclide Migration Studies

The deep-sea sediments with which most of our geochemical studies are being conducted are from a site designated PAC II about 1000 km north of Hawaii. This generic study region, previously referred to as Mid-Plate, Mid-Gyre I (MPG-I), was chosen because it meets the following site-selection criteria:

1. Tectonic stability. Areas chosen for study must be remote from lithospheric plate boundaries (mid-ocean ridges, subduction zones, and transverse fracture zones). Moreover, sediments must be highly predictable, both laterally and vertically. Predictability facilitates modeling the response of sediments to the perturbations that would be caused by a repository. Continuous sedimentation is an essential prerequisite for predictability. At PAC II the best dating techniques available (geomagnetic and ichthyolith stratigraphy) indicate that sedimentation has been

Table 1 Participators in the Seabed Working Group of the Nuclear Energy Agency, Organization for Economic Cooperation and Development

Seabed Working Group

NEA Secretariat: B. Ruegger
Task Group Coordinator: D. R. Anderson, United States

Canada	**G. Vilks**	**United Kingdom**	**F. S. Feates**
Japan	**R. Ichikawa**	**United States**	**D. Glenn Boyer (Chairman)**
France	**A. Barbreau**	**Commission of the European Communities**	**C. N. Murray**
Netherlands	**B. P. Hageman**		

Task Group	Canada	Japan	France	Netherlands	United Kingdom	United States	Commission of the European Communities
Physical ocean		Horibe	Madelain		H. Hill	Robinson*	
Waste form and canister			Pottier		Marsh	Magnani*	Lanza
Biology	Hargrave	Ichikawa	Belot	Cadee	Pentreath*	Yayanos	Schulte
Sediment and rock	Cranston		Rancon*	Kuijpers	Francis	Heath	Avogadro
Site criteria	Buckley	Hotta	Wannesson	Schuettenhelm	Searle*	Laine	
System analysis	Vilks	Ishikawa	de Marsily*	Van Weers	M. Hill	Marietta	Murray
Emplacement engineering			Boulanger	Nieuwenhuis	St. John	Talbert*	Murry

**Lead correspondent.*

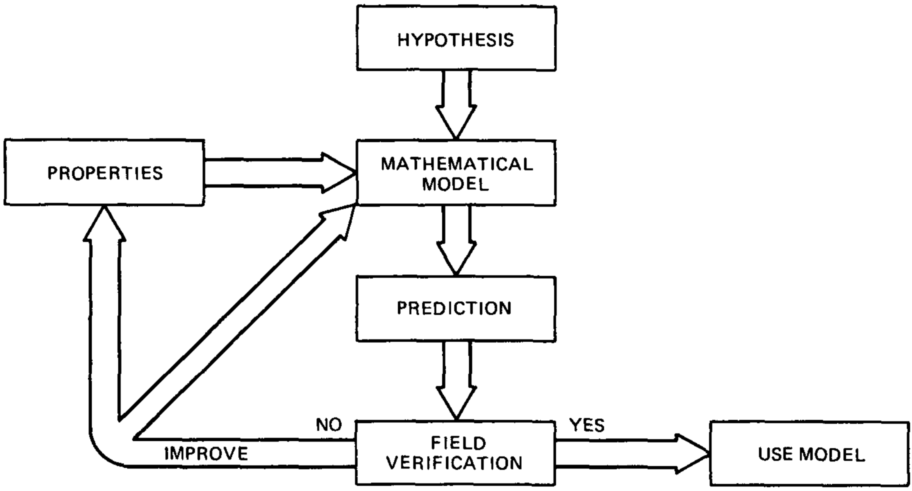

Figure 1 Research approach of the U. S. Subseabed Disposal Program.

continuous for roughly 60 million yr (see Fig. 2). These considerations imply that the integrity of a subseabed repository can be guaranteed, with a degree of certainty that is high relative to that for mined repositories, for tens of millions of years.

2. Location under the center of the north Pacific gyre (circulatory current regime). It was recognized early in the program that the water column is not suitable as a primary barrier to radionuclide migration. Nevertheless, location of a repository under the center of the north Pacific gyre would inhibit radionuclide migration in the event of an accident (failure of a canister to emplace properly, early canister failure, etc.).

3. Low biological productivity. In terms of organic carbon production, the PAC II study region is an oceanic analogue of the terrestrial deserts (see Fig. 3). In the event of an accident at PAC II, damage to the marine environment would be relatively low, and the transfer of radionuclides back to man would be inhibited.

4. Low resource potential. The PAC II study region is remote from marine fisheries, shipping lanes, and transoceanic cables. There are virtually no potential hydrocarbon resources here. Moreover, the manganese nodules in this area are low in copper and nickel, the primary elements of interest to potential deep-sea miners. These factors and the water depth at PAC II (>5000 m) ensure that no future society would be tempted to violate the integrity of a subseabed repository. Furthermore, no society less advanced than our own would be able to do so.

The sediments around a canister emplaced in a hypothetical subseabed repository have been divided into near- and far-field regions. The near field is defined as the region in which the maximum temperature exceeds 100°C. It will be shown that the volume of the near field would be small; the maximum extent of the 100°C isothermal surface would be about 1 m from the canister. (The regions affected by neighboring canisters would not overlap in a subseabed repository.) The near field is not, therefore, viewed as a significant bar-

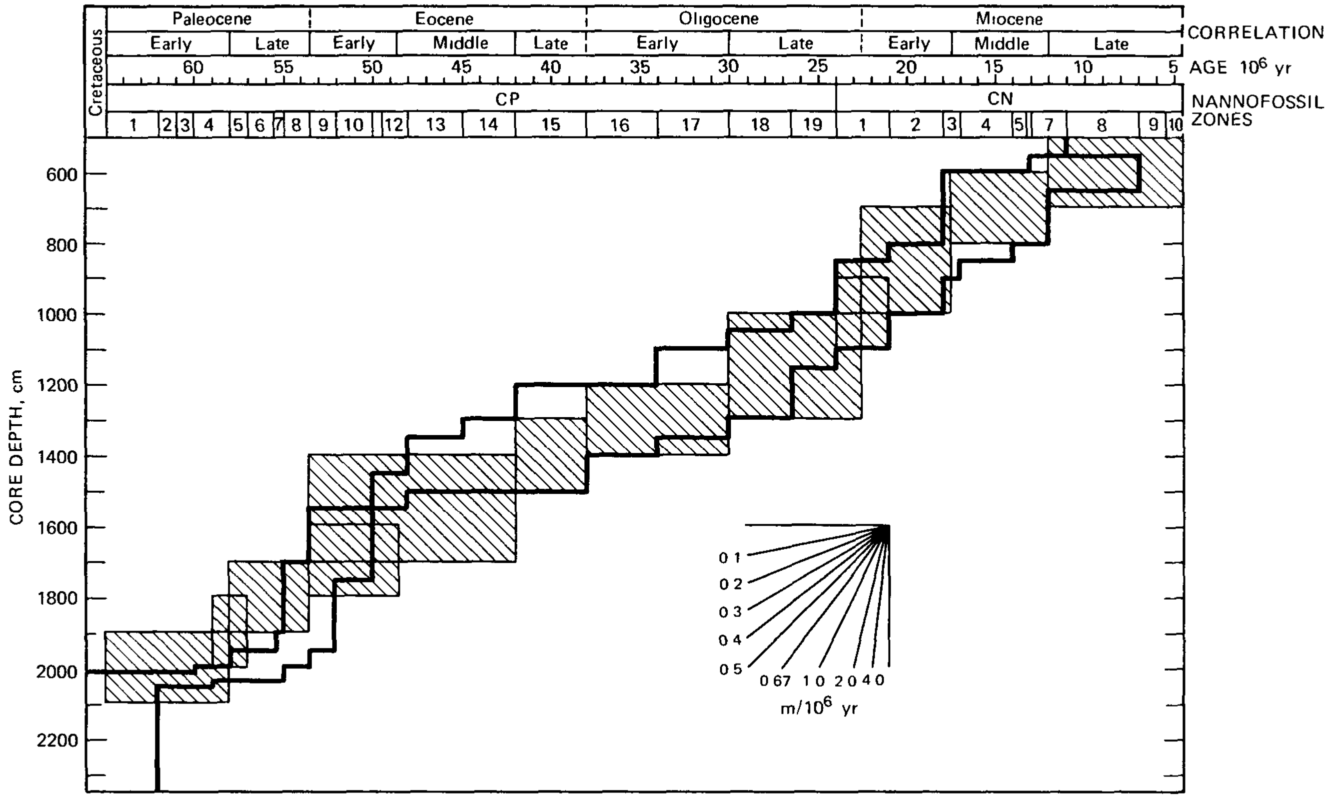

Figure 2 Age vs. depth relationship for Giant Piston Core 3, PAC II, study region. Hatched areas and heavy lines indicate uncertainties for initial and later studies, respectively.

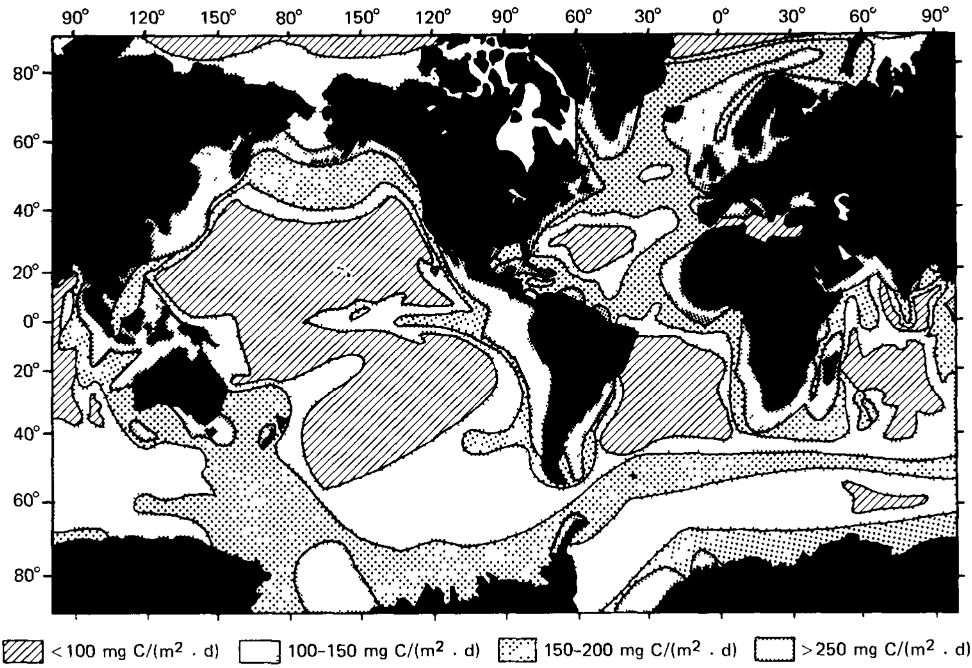

Figure 3 Distribution of primary carbon production in the world's oceans.

rier to radionuclide migration, even in the event of early failure of the waste package. Nevertheless, an intensive research program is being carried out to ensure that no process in the near field violates the integrity of the primary barriers (waste package and far-field sediments).

W. E. Seyfried of the University of Minnesota is studying sediment-seawater interactions under near-field conditions. At PAC II the sediments are saturated with pore water that does not differ significantly from normal seawater. The results of Seyfried and co-workers (Seyfried and Janecky, 1979; Seyfried, Thornton, and Janecky, 1981; Seyfried and Thornton, 1981), obtained using a modified Dickson-type hydrothermal apparatus, implied that smectite, the dominant mineral in subsurface PAC II sediments, is stable under near-field conditions. Some amorphous material initially present in the sediments crystallized as smectite during 1-month runs. These studies also demonstrated that, along with other changes, seawater becomes acidic on heating (see Fig. 4). This is caused by the precipitation of magnesium hydroxysulfate and smectite. Because this effect is strongly temperature dependent (see Fig. 5), canister lifetime considerations currently constrain the maximum temperature at the canister-sediment interface to values below 250°C. This temperature dependence also suggests that acidification would be confined to a small region surrounding each canister. Seyfried and co-workers are developing the capability of predicting quantitatively the effects of initial sediment composition on redox conditions during the thermal period, the first 300 to 500 yr after emplacement. Because the PAC II sediments are red clays, near-

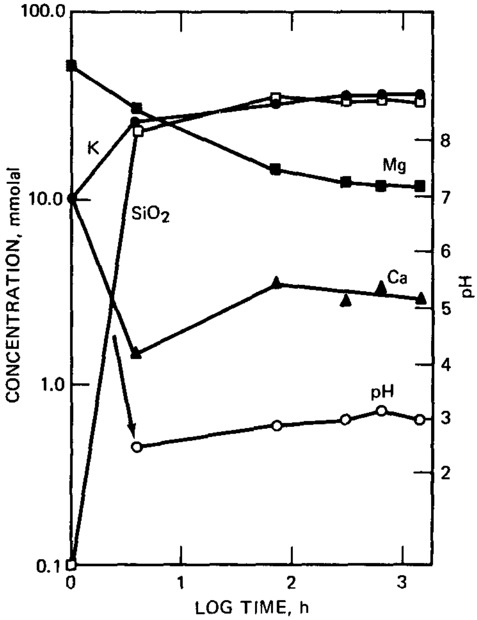

Figure 4 Chemical changes induced in seawater by heating to 300°C at 500 bars in contact with smectitic PAC II sediments.

field (and far-field) conditions would be relatively oxidizing.

Seyfried and co-workers are also beginning to study the effects of a thermal gradient on near-field chemistry (previous experiments were conducted at 200 or 300°C and 500 bars). In addition, programs are being developed at Sandia National Laboratories to evaluate the effects of radiation on near-field processes. Eventually the results of these experimental studies will be incorporated into computational models, such as the EQ3/6 software package (Wolery, 1979), to better quantify predictions of long-term near-field behavior. These predictions will, in turn, be tested during the In Situ Heat Transfer Experiment (ISHTE), described here. Nevertheless, the results of all work to date indicate that conditions in the near-field would neither preclude a 1000-yr period of waste package integrity nor jeopardize the ability of the far-field sediments to sorb radionuclides thereafter.

Extensive studies of the sorption of fission products and actinides by smectitic red clays from PAC II under far-field (<100°C) conditions have been carried out with batch equilibrium techniques by Erickson (1979a, 1979b, 1981a, 1981b), Heath and co-workers (Heath et al., 1979a, 1979b; Heath, Epstein, and Prince, 1977; Heath, Leinen, and Epstein, 1981), and Kenna (1981). These investigators examined the effects of sediment mineralogy, solution composition, temperature, time, and direction of reaction (sorption vs. desorption) on the radionuclide distribution coefficients. Figures 6, 7, and 8 show typical results of these studies.

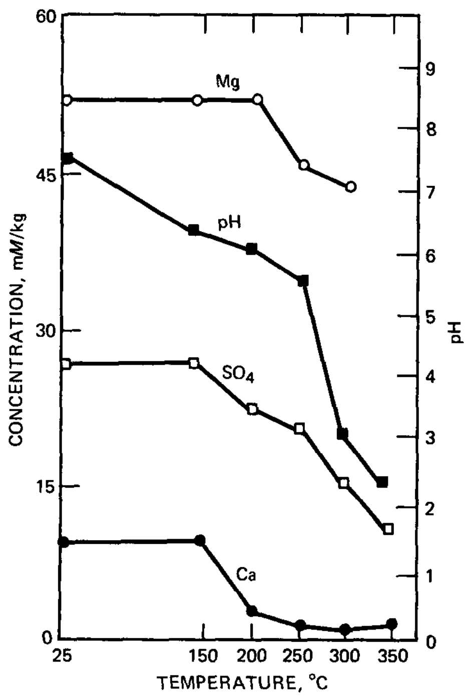

Figure 5 Chemical changes induced in seawater by heating to various temperatures.

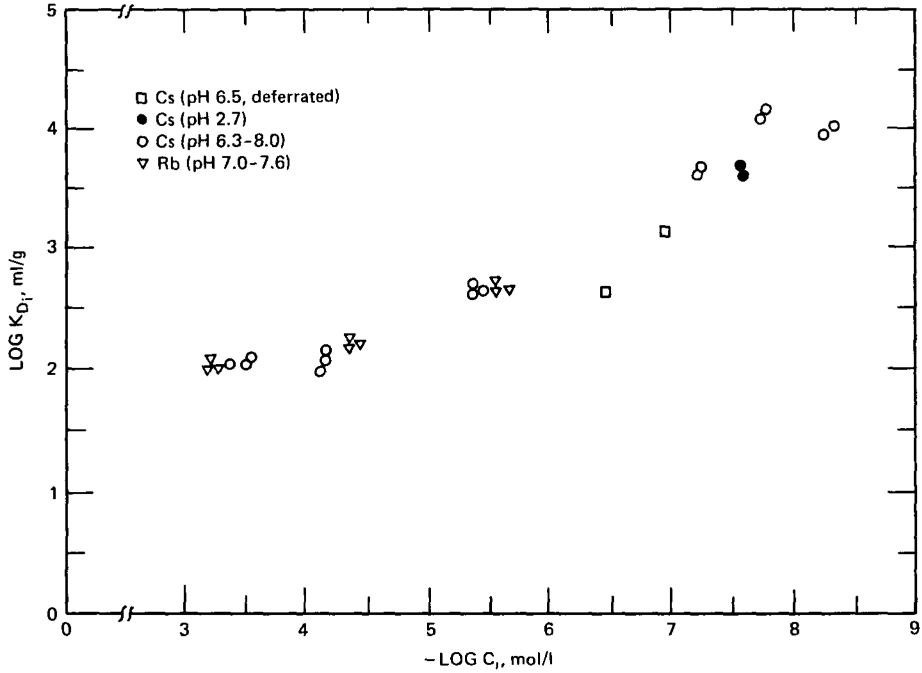

Figure 6 Distribution coefficients for cesium and rubidium between smectitic PAC II sediments and 0.68 NaCl solutions as a function of their final concentrations in solution.

In addition, the results of the sorption studies are currently being checked with column diffusion experiments (see Erickson, 1981b; Heath Leinen, and Epstein, 1981; Schreiner, Fried, and Friedman, 1981). Preliminary results agree well with the batch K_D experiments and imply that batch equilibrium experiments with deep-sea sediments can be used to obtain data for ion transport modeling.

The computer code IONMIG, developed at Sandia National Laboratories by Russo (1980) to model the migration of radionuclides through undisturbed deep-sea sediments, uses a two-dimensional, axisymmetric, explicit finite difference formulation. The heat generated by the 10-yr-old high-level waste in the cylindrical canister (3 m long and 0.3 m in diameter) is assumed to be 1.5 kw at the time of emplacement and to decay according to a schedule given by the ORIGEN code (Bell, 1973). The thermal (Fig. 9) and convective (Fig. 10) responses of the sediment-pore-water system* are calculated by the finite element code MARIAH (Gartling and Hickox, 1980). Clearly, MARIAH predicts that convective movement of pore waters through the sediments would be insignificant (<1 m during the thermal period); heat would be transferred mainly by conduction. This is a consequence of the low permeability ($\sim 10^{-7}$ cm/s for a hydraulic gradient of unity) of these fine,

As mentioned earlier, the regions of the sediments affected by each canister would not overlap in a subseabed repository.

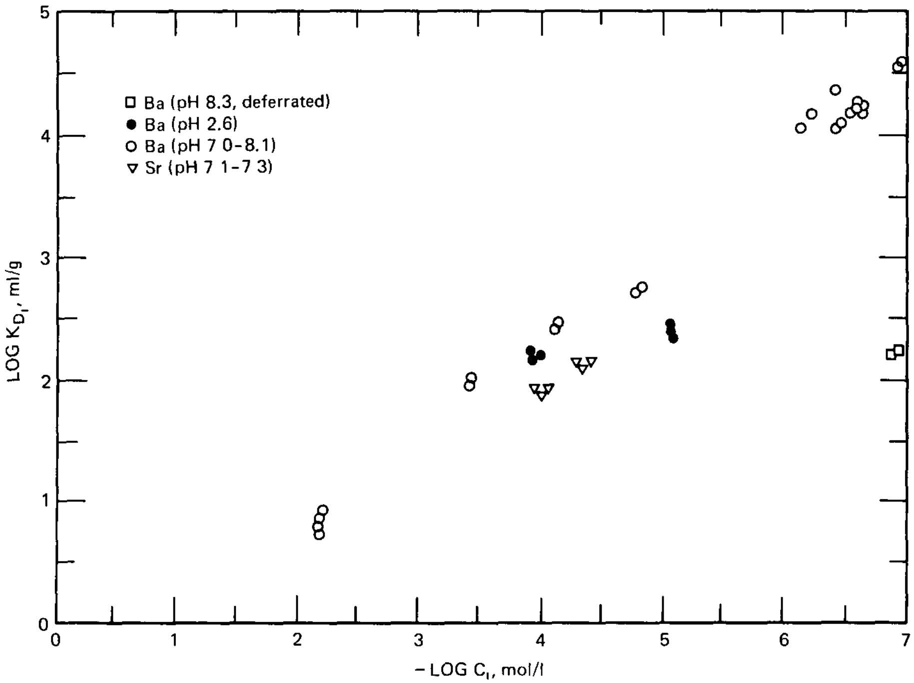

Figure 7 Distribution coefficients for barium and strontium between smectitic PAC II sediments and 0.68N NaCl solutions as a function of their final concentrations in solution.

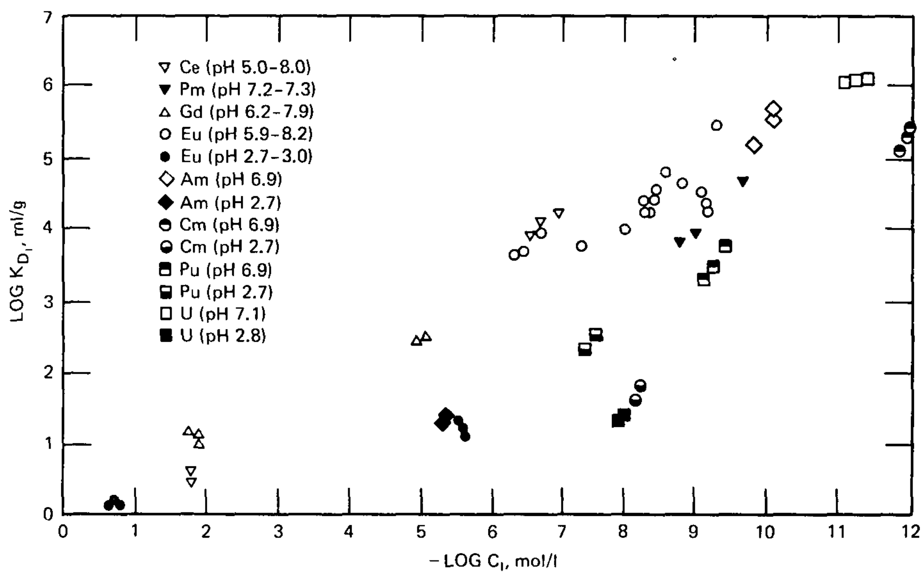

Figure 8 Distribution coefficients for curium, promethium, gadolinium, europium, americium, plutonium, and uranium between smectitic PAC II sediments and 0.68N NaCl solutions as a function of their final concentrations in solution.

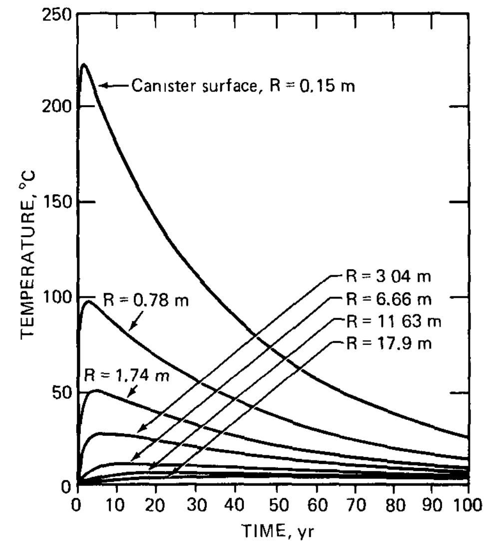

Figure 9 Temperature at seven locations radially outward from the center of a canister as a function of time (10-yr-old commercial high-level waste with initial loading of 1.5 kW).

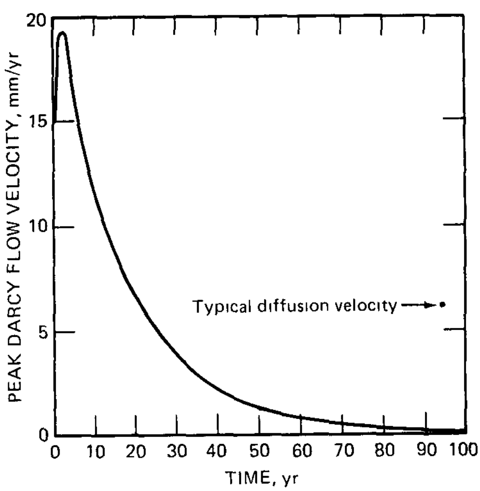

Figure 10 Maximum pore fluid velocity produced by a waste canister buried at a depth of 30 m.

pelagic clays (average grain size is 1μ or less). Because the tectonic stability of the PAC II site and the plastic nature of the sediments rule out the possibility of fracturing, radionuclide transport in these undisturbed sediments would take place only by molecular diffusion.

Preliminary modeling of the migration of several radionuclides in oxidized, smectitic sediments has been completed. The results, which are conservative because instantaneous release from the waste package after emplacement (30 m deep) was assumed,* are, nevertheless, very encouraging for fission products with moderate K_D values (10^2 to 10^5 ml/g), and for actinides with high K_D values (10^3 to 10^6 ml/g). These species, which include ^{137}Cs, ^{90}Sr, and ^{239}Pu, decay to innocuous levels before migrating significant distances in the sediments (see Fig. 11).

The results for some of the anionic species that have low K_D values (<1) in oxidized sediments are less encouraging (see Fig. 12). Efforts are currently under way to determine the extent of sorption of technetium by reduced sediments from other study areas. If these sediments do, in fact, immobilize Tc, a subseabed repository could be sited in an area in which, for example, reduced sediments overlie oxidized sediments, or technetium could be removed from high-level waste and buried in highly reduced sediments elsewhere.

**An intensive canister-overpack research and development effort is underway at Sandia National Laboratories in support of the SDP and the Waste Isolation Pilot Plant Project. Although much work remains to be done, it is expected that a titanium-based overpack capable of isolating the waste form for 1000 yr can be developed. Moreover, ongoing work at other laboratories is expected to produce a waste form capable of ensuring slow release of radionuclides to the sediments after the 1000-yr period of canister-overpack integrity.*

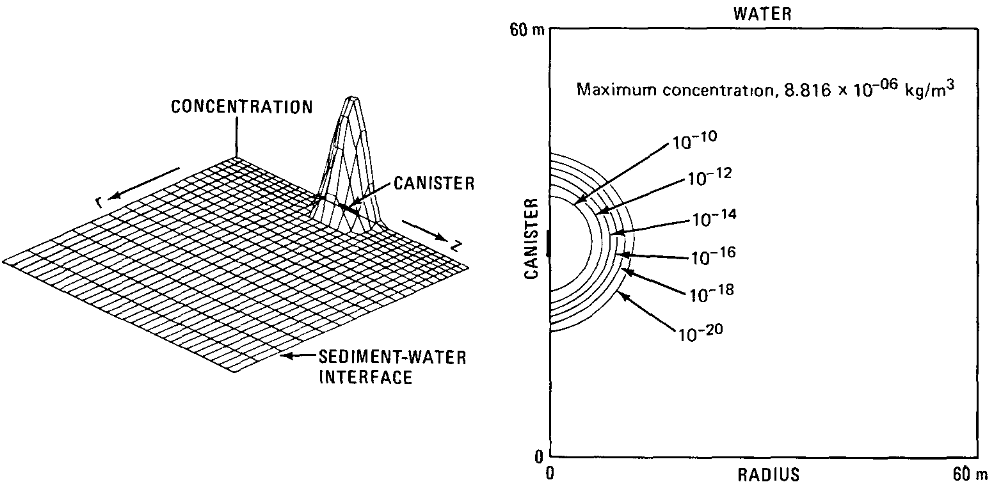

Figure 11 Plutonium concentration in the sediments at 100,000 yr. Instantaneous release is assumed.

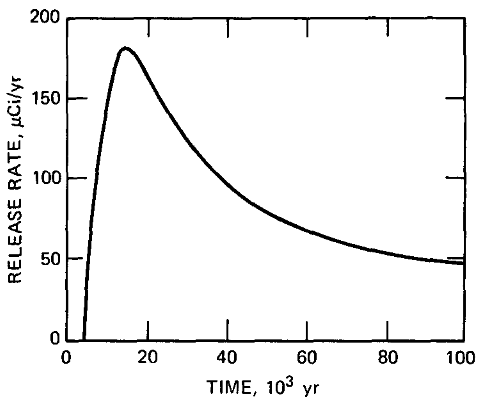

Figure 12 Release rate of ^{99}Tc from the sediment surface over a single canister buried 30 m deep. Instantaneous release is assumed.

Planning is under way for field testing of these laboratory measurements and computer modeling of radionuclide transport. The field test will be part of the ISHTE, a year-long experiment that will be deployed on the floor of the north Pacific in 1984 (see Percival et al., in press). Chemical tracers will be used to verify laboratory sorption and diffusion studies, as well as to determine the magnitude of any convection induced by the heat source. Thermal measurements will also be used to monitor convective flow. In addition, pore-water samples will be taken from the hot zone of ISHTE to verify the results of Seyfried and co-workers, and sediments will be analyzed after ISHTE is recovered.

Repository Design Concepts and Emplacement Options

The modeling discussed is based on a scenario in which canisters, one per hole, are placed 30 m deep in sediments with a total thickness of 60 m. Basaltic crust underlies the sediments. Canisters would be placed far enough apart so that the thermally and chemically affected zones of the sediments would not overlap; a grid with canister spacings of a few tens of meters would be more than sufficient.* This

**In a subseabed repository, the spacings between canisters would probably have only a very minor impact on the total cost of the repository.*

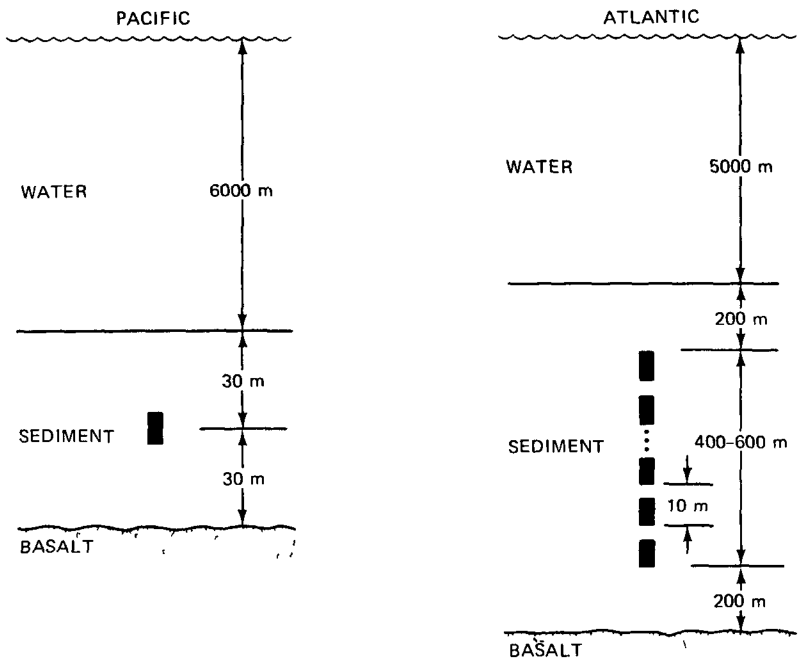

Figure 13 Base-case scenarios for emplacement of high-level waste canisters.

conceptual design (see Fig. 13) is applicable to the Pacific study region described.

In the Atlantic, a smaller ocean with higher sedimentation rates and thicker sediments, it would probably be feasible to place several canisters in each hole (see Fig. 13). Chemical modeling studies using an Atlantic-type conceptual design have not been carried out yet.

Several emplacement options appear feasible at this time. Two of these, penetration and drilled emplacement, are shown in Figs. 14 and 15, respectively.

The penetrators shown in parts a, c, and d of Fig. 14 would operate in a free-fall mode. Terminal velocities of 40 to 50 m/s would be attained during free fall even if the canister were lowered by wire before release (part c). It is important to note that the technology shown in part c is currently available. Free-fall penetrators could potentially reach burial depths of 20 to 30 m.

If ongoing radionuclide migration studies dictate that greater burial depths are necessary, boosted penetrators could be used (parts b, d, and e of Fig. 14). Canister velocities up to 100 m/s could probably be attained in the boosted mode.

Preliminary calculations indicate that dynamic rebound and plastic creep of sediments would probably result in closure along most of the hole immediately or soon after emplacement (Talbert, 1981a). Field testing will be required to verify these calculations.

Drilled emplacement is illustrated in Fig. 15. This technology is currently being used by the D/V *Glomar Challenger*. Figure 15 shows (a) the dril-

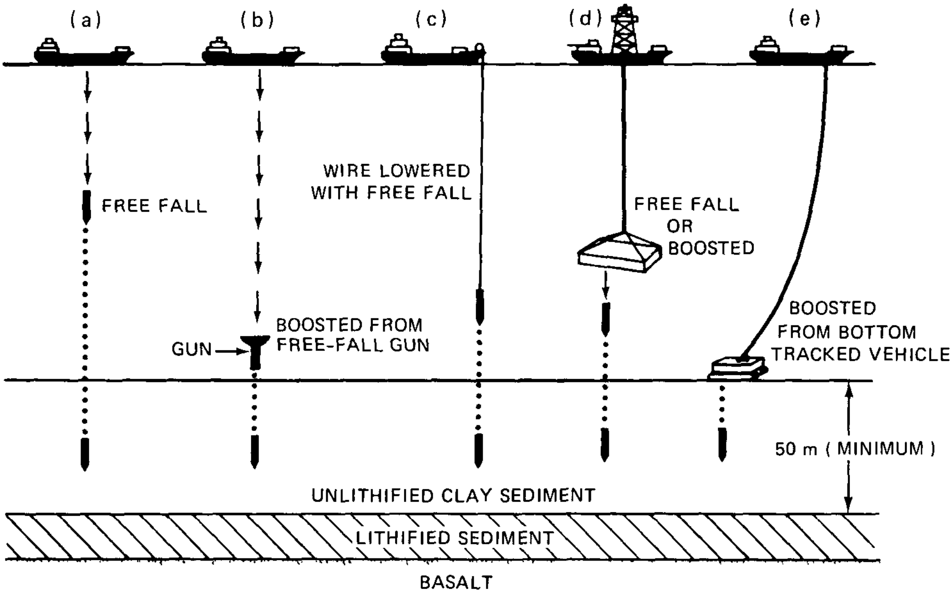

Figure 14 Penetrator emplacement concepts.

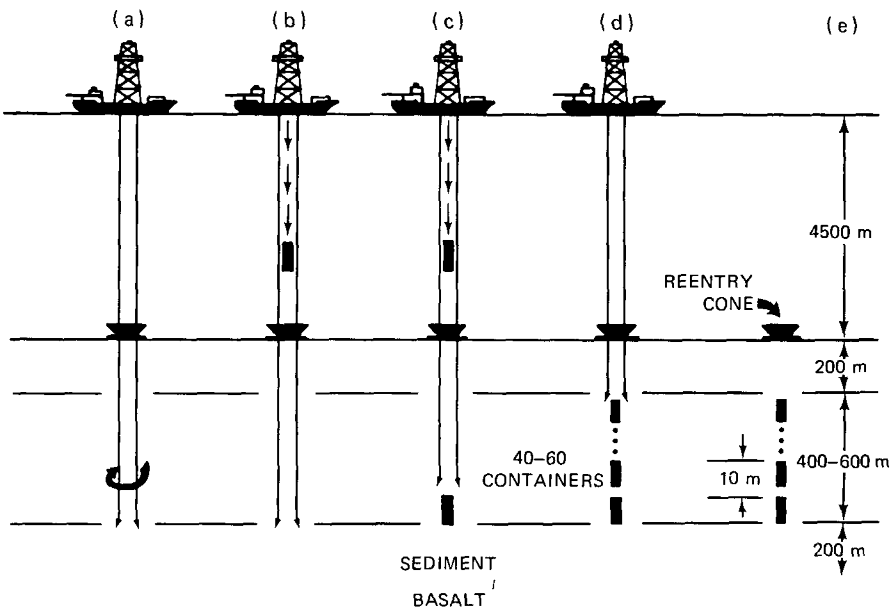

Figure 15 Drilled emplacement concepts.

ling of an emplacement hole; (b) emplacement of a canister at the end of a drilling pipe; (c) partial pipe withdrawal and backfill and reconsolidation with local sedimentary material; (d) emplacement of additional canisters; and (e) the final configuration, with all the canisters separated from each other and from the overlying water column by backfilled and reconsolidated sedimentary material.* Because of the time that would be required to drill each hole, this option would require that several canisters be placed in each hole and would most likely be applicable to the thick sediments of the Atlantic.

In addition to the emplacement options discussed here, trenching or some form of rammed injection might also prove feasible.

For any of these emplacement options, currently available navigational techniques could position the ship and the canister within meters of the desired location. Determining canister location after emplacement (or failure to emplace) is also feasible with current technology. Retrieval, too, is probably feasible with current technology, and, although it would be expensive and time consuming, costs would probably be comparable to retrieval from mined repositories.

The engineering studies that will be required by the SDP are discussed in greater detail by Talbert (1980).

Conclusion

Although much more work is needed to verify the technical and environmental feasibility of the subseabed disposal concept, all results to date indicate that this is a viable waste disposal option.

The sediments that would be used as backfill and, in fact, those which would surround the repository are similar mineralogically and chemically to the bentonites that are being considered for use as backfill in mined repositories

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Part III Waste Package Development and Design

The Use of Simple Transport Equations to Estimate Waste Package Performance Requirements

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A method of developing waste package performance requirements for specific nuclides is described. The method is based on:

- **Federal regulations concerning permissible concentrations in solution at the point of discharge to the accessible environment**
- **A simple and conservative transport model**
- **Baseline and potential worst-case release scenarios**

Use of the transport model enables calculation of maximum permissible release rates within a repository in basalt for each of the scenarios. The maximum permissible release rates correspond to performance requirements for the engineered barrier system.

The repository was assumed to be constructed in a basalt layer. For the cases considered, including a well drilled into an aquifer 1750 m from the repository center, little significant advantage is obtained from a 1000-yr as opposed to a 100-yr waste package. A 1000-yr waste package is of importance only for nuclides with half-lives much less than 100 yr which travel to the accessible environment in much less than 1000 yr. Such short travel times are extremely unlikely for a mined repository.

Among the actinides, the most stringent maximum permissible release rates are for ^{236}U and ^{234}U. A simple solubility calculation suggests, however, that these performance requirements can be readily met by the engineered barrier system. Under the reducing conditions likely to occur in a repository located in basalt, uranium would be sufficiently insoluble that no solution could contain more than about 0.01% of the maximum permissible concentration at saturation.

The performance requirements derived from the one-dimensional modeling approach are conservative by at least one to two orders of magnitude. More quantitative three-dimensional modeling at specific sites should enable relaxation of the performance criteria derived in this study.

Introduction

The long-term storage of high-level nuclear wastes in geologic formations requires a conservative approach to the design of the waste package and the nuclear waste repository. This conservatism is necessary because of the degree of uncertainty in the rates and manner in which radionuclides might be released from the repository and transported to the accessible environment. The current philosophy of the National Waste Terminal Storage program is to minimize such uncertainties by using a system of multiple engineered barriers around the waste form within the repository. These barriers, e.g., waste form, canister, buffer, overpack, and backfill (Fig. 1), will act in concert to inhibit the release of radionuclides from the waste

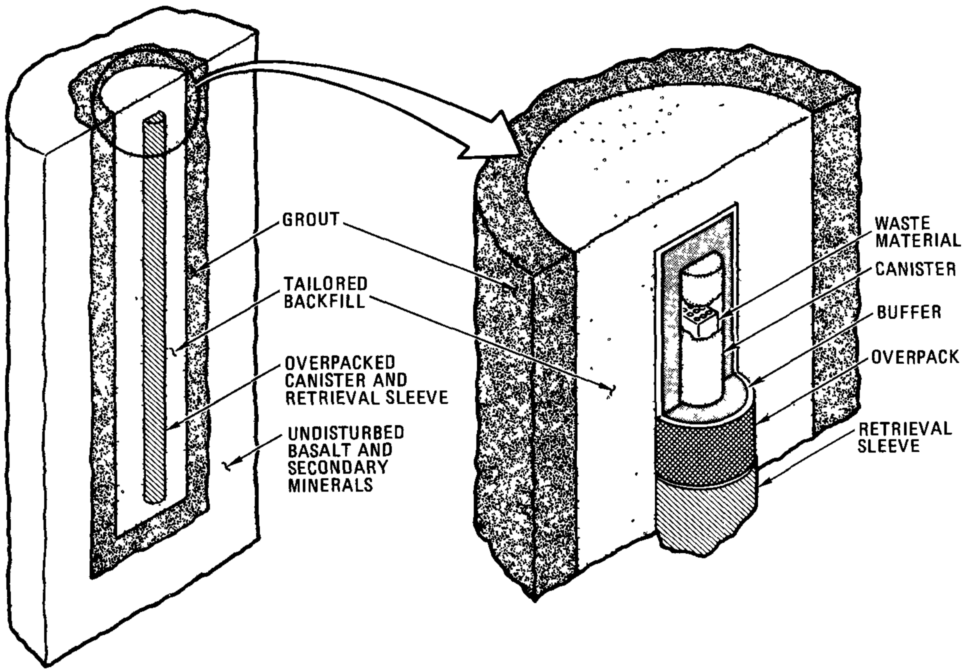


Figure 1 Engineered barrier system emplacement concept.

package. The functions of the different barriers envisaged by Coons et al. (1980) involve redundancy to prevent or reduce the effects of premature failure of one or more components of the system (Fig. 1).

The extent of multiple redundancy in the conceptual waste package is indicated in Table 1 (Coons et al., 1980). After emplacement in the repository, the function of the waste form is to retain the radionuclides; whereas the functions of the canister and remaining engineered barriers vary with time. During the thermal period (approximately the first 1000 yr of repository life), the canister, overpack, and backfill retard or prevent groundwater intrusion to the waste form, with the overpack functioning as the primary physical barrier and the backfill and canister functioning in supplementary roles. If the overpack fails prematurely, the buffer modifies groundwater chemistry to minimize corrosion of the canister. If the canister is also breached, the backfill operates for a time as the primary chemical barrier to radionuclide migration, with the basalt geology acting as a redundant chemical barrier. Canisters that survive the thermal period will continue to function as primary physical barriers to ingress of groundwater during the period of geologic control. After the canisters begin to fail, the basalt will function as the primary chemical barrier to radionuclide migration, with the backfill acting as a secondary barrier.

The preceding paragraph summarizes the approach taken by Coons et al. (1980) in defining the functions of an engineered barrier system throughout the life of a repository in basalt. These investigators did not, however, make any attempt to estimate performance requirements for the engineered barrier system other than to emphasize that it must meet stringent regulatory criteria (Nuclear

Table 1 Barrier Function Vs. Time

Item	Barrier	Operating period	Function
1	Geology (basalt)	Thermal period*	Supplementary chemical barrier to radionuclide migration
		Geologic control†	Primary chemical barrier to radionuclide migration
		Repository life‡	Physical isolation of waste material from man
2	Backfill	Thermal period	Primary chemical barrier to radionuclide migration Inhibitor of groundwater intrusion
		Geologic control	Secondary chemical barrier
3	Overpack	Thermal period	Primary physical barrier to groundwater intrusion Aid to retrievability
4	Buffer	Thermal period	Chemical inhibitor of canister corrosion if overpack fails
5	Canister	Pre-emplacement§	Physical support and protection for waste form
		Thermal period	Supplement to overpack in preventing groundwater intrusion Retrievability
		Geologic control	Primary physical barrier to groundwater intrusion
6	Waste form	Pre-emplacement and repository life	Retarder of radionuclides release if containment fails

**Thermal period is the time before 1000 yr of operation.*
†Geologic control is the time after 1000 yr of operation.
‡Repository life is the thermal period plus geologic control.
§Pre-emplacement is the time from canister filling to emplacement in the repository.

Regulatory Commission, 1980). This paper describes a simple approach to determining performance requirements for the engineered barrier system. The approach is a combination of data obtained from a one-dimensional transport model and information on the thermodynamic properties of key radionuclides under likely repository conditions.

Waste Package Performance Outline

The Nuclear Regulatory Commission (1980) established criteria for the long-term performance of the waste package throughout the life of a geologic repository. The relevant parts of 10CFR part 60, subpart E, are:

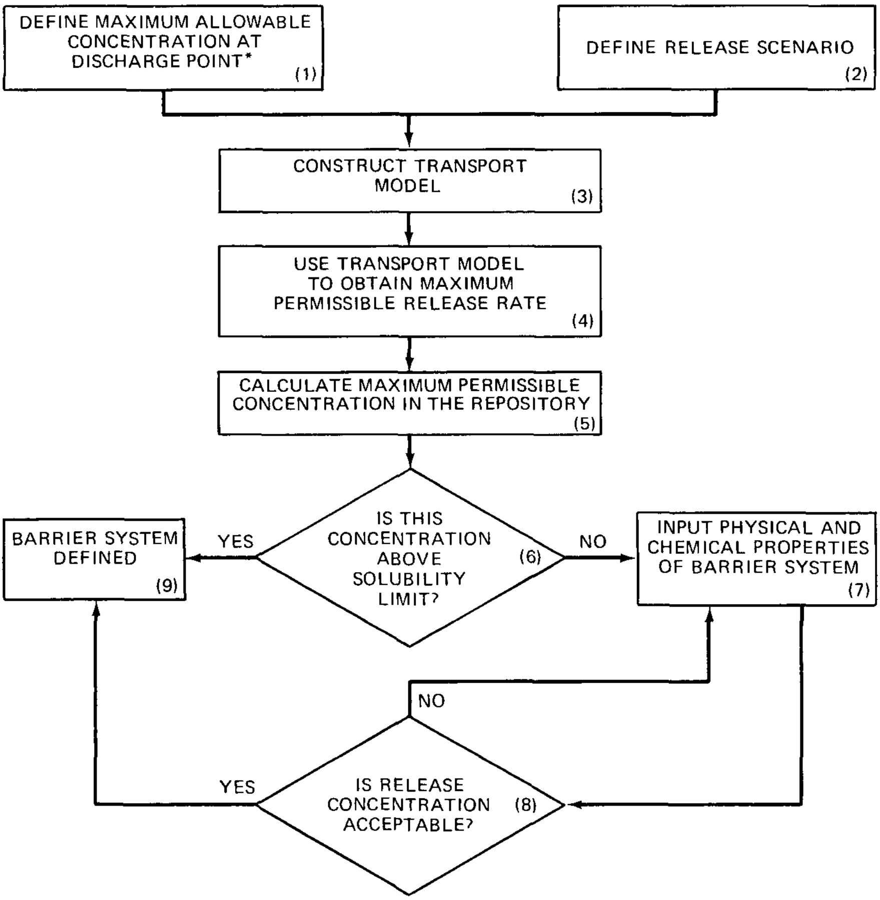
1. The Department (of Energy*) shall design waste packages so that there is reasonable assurance that radionuclides will be contained for at least the first 1000 years after decommissioning and for as long thereafter as is reasonably achievable given expected processes and events as well as various water flow conditions, including full or partial saturation of the underground facility.

2. 1000 years after decommissioning of the geologic repository operations area the radionuclides present in HLW (high-level waste) will be released from the underground facility at a rate that is as low as reasonably achievable and is in no case greater than an annual rate of one part in one hundred thousand of the total activity present in HLW within the underground facility assuming expected processes and events.

These criteria, while clearly defined, do not identify specific nuclides requiring particular attention by engineers developing the waste package. There is, for example, no attempt to relate the second criterion to the inventories of the different nuclides (which change with time) or to the widely varying hazards presented by different nuclides in terms of toxicity. This paper describes a method of estimating maximum permissible release rates of specific nuclides from the waste package and, therefore, establishing performance criteria for the waste package on a more quantitative basis. The approach, which is illustrated schematically in Fig. 2, can be summarized as follows:

- Step 1. The first step is to define a maximum allowable concentration of the radionuclide at the discharge point into the biosphere. In this paper the maximum permissible concentrations are taken to be the recommended concentration guides (RCG) established by the Nuclear Regulatory Commission (1976) for uncontrolled discharges of radioisotopes. The values will presumably change with time, but new discharge criteria can be easily incorporated.
- Step 2. A wide number of possible release scenarios can be envisaged. The simplest, of course, is release into the basalt and transport through this rock to the biosphere. A potentially more hazardous case might involve leakage up a fault plane or borehole plug into an aquifer system and discharge at a well drilled into the aquifer near the repository.
- Step 3. Transport from the repository to the biosphere has been treated in one dimension by using constant groundwater velocity. Although more complex two- and three-dimensional models will improve the precision of the transport analysis, the one-dimensional approach has a number of advantages. For example, it is inherently conservative because it ignores dispersion and dilution of the radionuclide in the plane at right angles to the direction of flow. Thus the long-term hazards posed by the waste are overestimated rather than underestimated. In addition, analytical solutions are available for some boundary conditions. These solutions can be used for rapid analysis of the sensitivity of discharge rates to variation of the controlling parameters.
- Step 4. Using the one-dimensional transport equations enables us to define the maximum permissible release rates consistent with items 1 and 2 in Fig. 2. A much more sophisticated model will, of course, be required for precise modeling of transport from the repository and for licensing proceedings. The model used here was deliberately adopted and applied in such a way that a more detailed approach will lead to increased (i.e., less conservative) maximum permissible release rates.
- Step 5. If we know the release scenario, using the maximum per-

**Parentheses indicate author's inserts.*



***See Nuclear Regulatory Commission, 1976.**

Figure 2 Performance and functional criteria for engineered barrier system.

missible release rate, we can calculate the maximum allowable concentration of the radionuclide in the repository groundwater.

- Step 6. For some nuclides the maximum allowable concentration within the repository may be above the saturation limit for the stable compound of the element. In such cases the radionuclide should precipitate, leaving a saturated solution of lower concentration than the maximum allowable. If this occurs, the engineered barrier system would not have to function in any specific way for the nuclide of interest.
- Steps 7 and 8. If the maximum allowable concentration is below the saturation limit, physical and chemical barriers must be constructed for the nuclide of concern so that the maximum permissible release rate is not exceeded. The physical and chemical properties of the barriers must be altered by changing chemical composition, thickness, etc., until the release rate is reduced to an acceptable value.
- Step 9. Iteration and adjustment of barrier properties should result in waste package performance that leads to acceptable concentrations of

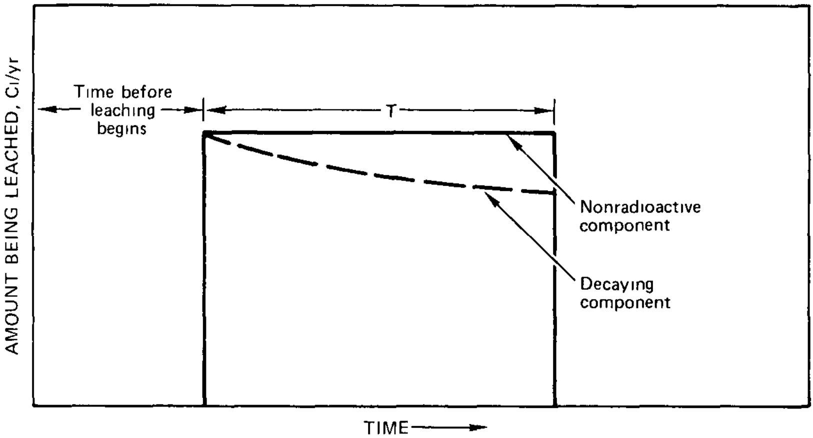


Figure 3 Schematic representation of radionuclide release model.

radionuclides at the discharge point. A more complete discussion of the approach to defining waste package performance requirements begins with the establishment of the transport model described here.

The transport equation in one dimension, corrected for radioactive decay, is

$$aV \frac{\partial^2 c}{\partial x^2} - V \frac{\partial c}{\partial x} - R \frac{\partial c}{\partial t} - R\lambda c = 0 \quad (1)$$

where a = dispersive length of the medium
V = water velocity
R = radionuclide retardation factor
c = radionuclide concentration
x = distance
t = time
λ = the radionuclide decay constant

Haderman (1980) solved this equation for this case of band radionuclide release (Fig. 3) and certain boundary conditions. Since the approach taken here is to define the *maximum* permissible concentration at the interface with the biosphere, solutions of Eq. 1 for all values of time are not required. The two important parameters required are: (1) the time after leaching begins when the discharge rate to the environment peaks and (2) the height of the peak at this time. If the geologic medium initially contains none of the radionuclide and if it is infinitely dilute at the boundary with the environment, the solution can be represented by

$$C_{x,t} = \frac{C_0}{2} (F) \exp(-\lambda t) \quad (2)$$

where $C_{x,t}$ is the concentration at distance x from the repository at time t, C_0 is the initial concentration, and F is a parameter that depends on time, distance, velocity, dispersion, and retardation; it is a sum of four error function complements, the exact form of which is omitted for simplicity. The interested reader is referred to Haderman (1980) for details.

When we inspect the form of F, we see that there are a number of simple relationships between the transport properties that govern its magnitude. First, there is a characteristic transport time, t_c, which is the time after leaching begins that the radionuclide discharge rate to the

Table 2 Values of F in Eq. 2 at Time of Peak Discharge (t_c)

	Distance/dispersive length (x/a)						
$(t_c - T)/t_c$	1	4	10	100	1000	10,000	10^5
0	1.43	1.26	1.17	1.06	1.02	1.01	1.00
0.5	0.45	0.80	1.01	1.06	1.02	1.01	1.00
0.7	0.22	0.43	0.64	1.05	1.02	1.01	1.00
0.9	0.06	0.13	0.19	0.48	1.01	1.01	1.00
0.95	0.03	0.06	0.09	0.29	0.75	1.01	1.00
0.99	0.005	0.011	0.016	0.058	0.18	0.52	0.98

environment peaks at distance x. The time of peak discharge is given by

$$t_c = \frac{xR}{V} \tag{3}$$

In Eq. 3 the retardation factor, R, for the nuclide of interest is the ratio of groundwater velocity to nuclide velocity. It is related to the physical and chemical properties of the rock through which transport is occurring by the expression

$$R = 1 + K_d \frac{e}{\phi} \tag{4}$$

where K_d is the ratio of radionuclide sorbed on the rock to that remaining in solution at equilibrium (ml/g), e is the density of the rock, and ϕ is its porosity. The characteristic travel time, which is different for each nuclide, is the ratio of distance x to nuclide velocity (V/R) (Eq. 3).

The factor F can be tabulated as a function of just two parameters at the time of peak discharge, t_c:

$$\frac{t_c - T}{t_c} \text{ and } \frac{x}{a} \tag{5}$$

where T is the overall time of leaching of the waste (Fig. 3). Values of F are tabulated in Table 2.

To obviate the need to look up the value of F for all values of travel time, distance, etc., we should note that there are two end-member approximations to Eq. 2 which apply for most values of the transport parameter. If the time of peak discharging concentration is t_c years after leaching begins, Eq. 2 can readily be transformed to a peak discharge rate at the biosphere (D) in curies per year (Ci/yr). This is done by noting that, for the step release function, the ratio of the peak discharging concentration ($C_{x,t}$) to the initial concentration in the repository (C_0) is the same as the ratio of D to the amount discharging from the canisters into the repository (D_0) (Ci/yr). Thus D_0 is the ratio of the repository inventory (I) in curies divided by total leach time (T). Rearranging Eq. 2 and substituting for D_0 yields either

$$D \simeq \frac{I}{2T} \exp(-\lambda t_c) \tag{6}$$

or

$$D = \frac{I}{4t_c} \sqrt{\frac{x}{a}} \exp(-\lambda t_c) \tag{7}$$

Equation 6 applies if total time of leaching (T) is of the same order as the characteristic travel time (t_c). Equation 7 applies if t_c is very much greater than T. The two cases are illustrated schematically in Fig. 4.

Equations 2, 6, and 7 can be applied to nuclides, such as fission products, which decay but are not formed during transport from the repository to the biosphere. These

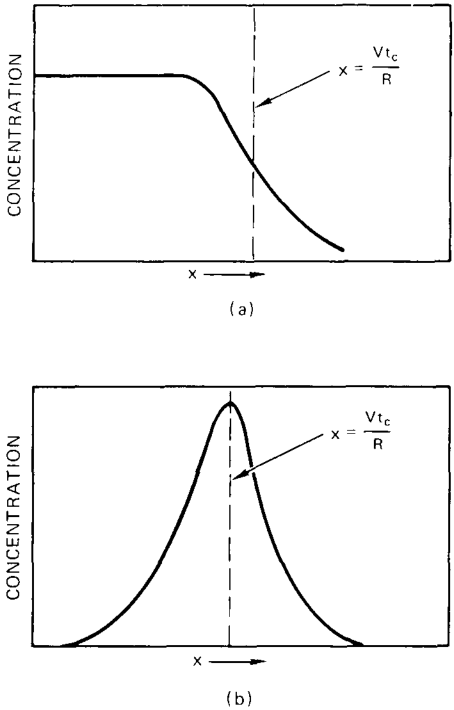


Figure 4 Schematic representation of the two end-member solutions to the transport equation. (a) Leach time ⩾ Travel time (Eq. 7). (b) Leach time ≪ Travel time (Eq. 8).

equations are not, however, strictly applicable to actinide chains whose members are generated and decay during the transport process, but there are a number of simplifications that enable the equations to be approximately applied to most members of actinide decay chains. For the purpose of making such calculations, I considered the members of actinide chains in three groups:

1. Low concentrations of all precursors
2. $t_{1/2} \ll t_c$; high concentrations of at least one precursor
3. $t_{1/2} \approx t_c$; high concentrations of at least one precursor

These three cases and the appropriate simplifications are considered in turn.

1. Nuclides that are at the beginning of decay chains or do not have a precursor present in high concentration in the waste form can be modeled by using Eqs. 2, 6, and 7 directly since little nuclide production takes place after the repository is closed or during transport to the biosphere. Examples of such radionuclides in spent irradiated fuel are ^{245}Cm, ^{238}U, and, after 10^3 yr, ^{237}Np.

2. If the half-life of a nuclide is very much shorter than its transport time, little of the initial inventory in the repository can be transported to the biosphere. If the radionuclide has a long-lived parent, however, there may be a significant discharge of the daughter because it is being generated during transport of the parent. In this case the characteristic transport time for the daughter (t_c) must be controlled by that of the parent. Thus the concentration of the daughter will peak at about the same time as that of the parent. This leads to the approximation:

$$t_{c,d} = t_{c,p} \quad (8)$$

where subscripts d and p refer to daughter and parent, respectively. Barney and Wood (1980) showed that, as a good approximation for short-lived daughters, the peak discharge rate of the daughter D_d at $t_{c,p}$ is related to D_p as follows:

$$D_d = D_p \frac{R_p}{R_d} \qquad \text{Ci/yr} \quad (9)$$

3. If the daughter has a long half-life, its time of peak discharge will depend on the concentrations of precursors and the relative half-lives of parent and daughter. Its characteristic transport time will in most cases be dominated by that of the parent but will be shifted either early or late because of the differences between migration rates of parent and daughter. An estimate of the magni-

Table 4 Parameters Used in Transport Calculations

Case 1	Path length (x), 25,000 m Dispersive length (a), 2.5 m Leach time (T), variable (1000-yr reference) Water velocity (V), 0.1 m/yr Porosity (ϕ), 0.01
Case 2	Path length (x), 25,000 m Dispersive length (a), 2.5 m Leach time (T), variable (1000-yr reference) Water velocity (V), 20 m/yr Porosity (ϕ), 0.05
Case 3	Path length (x), 2,500 m Dispersive length (a), 2.5 m Leach time (T), variable (1000-yr reference) Water velocity (V), 1000 m/yr Porosity (ϕ), 0.1
Repository dimensions	3000 by 2400 by 12.5 m in all three cases

ing basalt. Brief descriptions of the three reference cases follow.

Case 1. Transport from the repository to a river, a drinking water source, through a medium with low hydraulic conductivity (1 m/yr), low porosity (0.01), and hydraulic head gradient of 10^{-3}. The value of hydraulic conductivity is toward the high end of values for Pasco Basin basalt on the Hanford Site (Gephart et al., 1979), and the head gradient corresponds to measured values in the Mabton interbed.

Case 2. Tran^port from the repository to the river via release into a fractured interflow zone, with hydraulic conductivity of 10^3 m/yr, porosity of 0.05, and head gradient of 10^{-3}. The hydraulic conductivity in this case is an average value for Grande Ronde basalt flow contacts (Gephart et al., 1979).

Case 3. Leakage from the repository center into the unconfined aquifer (hydraulic conductivity, 10^5 m/yr), (Gephart et al., 1979) and transport to the surface through a well drilled 1750 m from the center of the repository in the direction of fluid flow. The porosity was taken to be 0.1.

In all three cases (detailed in Table 4), the calculations were performed under two assumptions: (1) Leaching begins after 1000 yr of repository life, and (2) leaching begins after 100 yr of repository life.

Sorption Data. To calculate transport times to the biosphere for the radionuclides in Table 3, we must know the retardation factors that are characteristic of each element for the specific environment under consideration. Wherever possible, retardation factors have been calculated from Eq. 4 using sorption distribution coefficients tabulated by Ames and McGarrah (1980) and Barney (in Smith et al., 1980) (Table 5). These distribution coefficients, which exhibit a fairly wide range, were measured between Hanford basalt and simulated Hanford groundwater. Values toward the low end of those measured between 23 and 150°C were adopted throughout (Table 5). Estimates were made for elements for which data are unavailable by drawing analogies with elements that have been studied (Table 5).

Table 5 Sorption Distribution Coefficients (K_d) for Basalt*

Element	Observed range	Estimated K_d	Value adopted
Cs	70–2100		70
Sr	58–240		60
Se	0–10† (17–80)		0
I	0–3		0
Tc	0–40† (4000)		0
Np	3–16† (150)		3
Ra	48–150		50
U	0–70†		0
Pu	$10\text{–}10^4$		50
Am	$10\text{–}10^5$		20
H		0	
C		0	
Kr		0	
Sm		100	
Cm		20	
Pa		50	
Th		50	
Ac		10	
Zr		10	
Pd		10	
Cd		10	
Sn		10	
Ho		10	
Pb		10	

**Data from Ames and McGarrah (1980) and Barney (in Smith et al., 1980)*

†Data obtained under oxidizing conditions Values in parentheses are for anoxic conditions ($fO_2 = 8 \times 10^{-7}$ atm)

Consider transport from the repository to a drinking water source 25 km away via flow through the unit discussed in Case 1. The mean water velocity through the medium is given by

$$V = \frac{C \cdot \nabla H}{\phi} = \frac{1 \times 10^{-3}}{10^{-2}} = 0.1 \quad \text{m/yr} \quad (11)$$

where C is the hydraulic conductivity, ∇H is the head gradient, and ϕ is the porosity. Thus a nonsorbing nuclide will take 250,000 yr to travel from the repository to the discharge point 25 km away.

The discharge rates of the nuclides depend on leach time only if the latter is within one to two orders of magnitude of the transport time (Eq. 7). A small value of dispersive length, a (2.5 m), was chosen to maximize the calculated discharge rates in Eq. 7 and the region over which Eq. 6 dominates. The discharge rates at the calculated peak times were then obtained from Eqs. 6 and 7. The actual concentrations in solution at the time of peak discharge depend, of course, on the volume of water passing through the repository. The repository dimensions used in the calculations are given in Table 4. The volume of water passing

Table 6 Nuclides Above Recommended Concentration Guide in Case 1 for 1000-yr Leach Time*

Nuclide	Time of peak discharge, yr	Rate of peak discharge, Ci/yr	RCG, μCi/l	Concentration in solution, μCi/l	Leach rate required to keep below RCG, yr^{-1}
^{129}I	$2\ 5 \times 10^{5}$	$1\ 5 \times 10^{1}$	6×10^{5}	5	2×10^{9}
^{79}Se	$2\ 5 \times 10^{5}$	$1\ 3 \times 10^{1}$	3×10^{3}	4	1×10^{7}
^{99}Tc	$2\ 5 \times 10^{5}$	27	2×10^{1}	9×10^{2}	4×10^{8}
^{210}Pb	$2\ 5 \times 10^{5}$	$7\ 2 \times 10^{4}$	10^{4}	2×10^{2}	8×10^{7}†
^{226}Ra	$2\ 5 \times 10^{5}$	$1\ 4 \times 10^{4}$	3×10^{5}	5×10^{3}	1×10^{6}†
^{230}Th	$2\ 5 \times 10^{5}$	$1\ 4 \times 10^{4}$	2×10^{3}	5×10^{3}	1×10^{4}†
^{234}U	$2\ 5 \times 10^{5}$	1 8	3×10^{2}	60	1×10^{7}
^{235}U	$2\ 5 \times 10^{5}$	9×10^{2}	3×10^{2}	3	2×10^{6}
^{236}U	$2\ 5 \times 10^{5}$	1 6	3×10^{2}	53	1×10^{7}
^{238}U	$2\ 5 \times 10^{5}$	1 5	4×10^{2}	50	2×10^{7}

**Case 1 discharge 25 km from the repository Velocity of water flow is 0 1 m/yr, and leach time is 1000 yr, leaching begins after 1000 yr*

†Leach rates of ^{234}U

through the repository in any year is given by

$$\text{Volume} = V \times \text{width} \times \text{height} \times \text{porosity} \quad (12)$$

Using the smaller value of width given in Table 4 yields a volume of 3×10^4 l/yr and maximizes the concentrations in solution.

Leaching was assumed to begin after either 100 or 1000 yr and the reference leach time was taken to be 1000 yr, i e., as soon as leaching begins, the rate of release is 1 part in 1000 (10^{-3}) per year. Given this reference release rate and the volume of water passing through the repository, we can calculate the concentrations of nuclides at the discharge point Table 6 lists the peak concentrations of nuclides in solution that would produce peak discharge rates above the RCG for uncontrolled discharges (Nuclear Regulatory Commission, 1976) at a release rate of 10^{-3}/yr in the repository. No allowance was made for any possible dilution in the biosphere. The reference release rate can, of course, be adjusted to any desired value. The rate of 10^{-3}/yr was chosen on a conservative basis because it is 100 times greater than the Nuclear Regulatory Commission's criterion (1980). The maximum rate of release allowable to keep the discharge concentration below the RCG is also given in Table 6. Note that the absolute discharges are very small and that dilution at the discharge point [e.g., 10^{12} to 10^{14} l/yr to Columbia River (Energy Research and Development Administration, 1975)] would produce concentrations many orders of magnitude below the RCG. Because the characteristic travel times are very long ($\geq$250,000 yr), Eq. 7 controls peak discharge rates for all rates greater than about 10^{-5}/yr. Thus release rate has little effect on peak discharge rate until it becomes less than 10^{-5}/yr.

Dilution Caused by Dispersion. Dispersion in the plane at right angles to the direction of flow will, of course, tend to reduce the concentrations of radionuclides in the discharg-

Table 7 Nuclides Above Recommended Concentration Guide in Case 2 for 1000-yr Leach Time*

Nuclide	Time of peak discharge, yr	Rate of peak discharge, Ci/yr	RCG, μCi/l	Concentration in solution, μCi/l	Leach rate required to keep below RCG, yr^{-1}
^{79}Se	1250	8	3×10^{3}	3×10^{1}	1×10^{5}
^{99}Tc	1250	3×10^{2}	2×10^{1}	10	2×10^{5}
^{107}Pd	$6 3 \times 10^{[illegible]}$	0 2	3×10^{3}	7×10^{3}	4×10^{5}
^{129}I	1250	8×10^{1}	$6 \times 10^{[illegible]}$	3×10^{2}	2×10^{6}
^{233}U	$1 9 \times 10^{[illegible]}$	6×10^{2}	3×10^{2}	21	2×10^{7}†
^{234}U	1250	18 5	3×10^{2}	8×10^{1}	4×10^{5}
^{236}U	1250	5 5	3×10^{2}	2×10^{1}	1×10^{4}
^{237}Np	$1 9 \times 10^{5}$	6 6	3×10^{3}	2×10^{1}	4×10^{6}
^{238}U	1250	7 5	4×10^{2}	3×10^{1}	2×10^{4}

**Case 2 discharge 25 km from the repository Velocity of water flow is 20 m/yr, and leach time is 1000 yr, leaching begins after 1000 yr*

†Leach rate of [illegible] Np

ing solutions. The approximate magnitude of this effect can be calculated from the characteristic dispersion distance, Z, at the discharge point:

$$Z \simeq \sqrt{ax} = 250 \text{ m}$$

Thus, in traveling from the repository to the discharge region, the nuclide disperses approximately 250 m in the directions at right angles to the direction of flow, assuming that the medium is homogeneous and the dispersive length is constant. This would result in the nuclide's being about 50 times more dilute at the discharge point than was calculated from the one-dimensional equation.

If the repository were directly connected to an interflow zone with fairly large hydraulic conductivity, several important differences in discharge time and rate would be observed. With the hydrologic properties listed in Table 6, a nonsorbing nuclide would travel with a velocity of 20 m/yr and would, therefore, begin to discharge 1250 yr after leaching begins. Thus, since travel times are relatively short, Eq. 6 will dominate for many more nuclides than those in Case 1, and discharge rates (Ci/yr) will be considerably higher. In addition, even if Eq. 7 dominates, t_c is 200 times smaller for all nuclides; thus the discharge rate will be 200 times larger for such cases. Dilution of the radionuclides in the rock column is, however, much greater than in Case 1 because of the faster flow rate and greater porosity. Thus the volume of water flowing through the repository is 3×10^7 l/yr, and, if Eq. 7 dominates, nuclides will be five times more dilute than in Case 1 despite the rapid transport to the biosphere (Table 7). If Eq. 6 dominates in both Cases 1 and 2, a nondecaying nuclide will be 10^3 times more dilute in Case 2 than in Case 1. This is reflected in the release rate required to keep below RCG in the discharging solution in the two cases. If we consider ^{129}I, which does not decay significantly during the longer transport time of 2.5×10^5 yr, then the leach rate to keep below RCG is 2×10^{-9} in Case 1 and 2×10^{-6} in Case 2. The rate of peak discharge, however, is slightly less in Case 1 for the reference leach time of 1000 yr.

The calculations have been performed both by assuming initial con-

Table 8 Nuclides Above Recommended Concentration Guide in Case 3 for 1000-yr Leach Time*

Nuclide	Time of peak discharge, yr	Rate of peak discharge,† Ci/yr	RCG, μCi/l	Concentration in solution, μCi/l	Leach rate required to keep below RCG, yr^{-1}
^{129}I	2 5	8×10^{1}	6×10^{5}	3×10^{4}	2×10^{4}
^{239}Pu	2500	3×10^{3}	5×10^{3}	1 0	2×10^{5}
^{240}Pu	2500	4×10^{3}	5×10^{3}	1 4	4×10^{6}
^{242}Pu	2500	19	5×10^{3}	6×10^{3}	8×10^{4}
^{241}Am	1000	9×10^{3}	4×10^{3}	3 0	1×10^{6}
^{243}Am	1000	157	4×10^{3}	4×10^{2}	1×10^{4}

**Case 3 leaching into an aquifer, exiting through a well drilled 1750 m from the repository center Leach time is 1000 yr, leaching begins after 100 yr*

†Assuming all the nuclide discharges at the well.

tainment for 1000 yr and by assuming that release begins after 100 yr. The results show no significant difference. We can conclude from Cases 1 and 2 that, for a path length of 25 km, the advantages to be gained from placing the repository in a medium with hydraulic conductivity of 10^0 as opposed to 10^3 m/yr are surprisingly small. The main advantage is the retardation of the peak discharge time. Great advantages in discharge rate can be obtained only if the hydraulic conductivity is so low that the characteristic transport time of the nuclide is much longer than its half-life.

To calculate a potentially hazardous case, I assumed that waste from half the repository leaks up a crack in, for example, a borehole plug in the center of the repository. The waste is transported immediately into an unconfined aquifer and reaches the surface through a well drilled 2500 m from the center of the half of the repository concerned (1750 m from the actual repository center). If the leakage begins 100 yr after the repository is closed, isotopes of plutonium and americium comprise the greatest hazards in the discharging solution (Table 8). (It was assumed that the volume of solution in which the nuclides are contained is given by Eq. 11, 3×10^9 l/yr.) In Cases 1 and 2, the actinides plutonium and americium do not constitute great hazards because decay drastically reduces their concentrations long before they reach the biosphere. In Case 3, however, the active short-lived nuclides reach the environment after relatively short times and are not adequately contained by the geologic media. The longer-lived nuclides, such as ^{79}Se, ^{99}Tc, and ^{237}Np, which comprise hazards in Cases 1 and 2, are less hazardous in Case 3 because of their greater dilution in the rapidly conducting aquifer. As with Cases 1 and 2, little advantage was found to be derived from containing the waste for 1000 yr instead of 100 yr. It should be noted that nonsorbing nuclides such as ^{129}I have extremely short travel times within the aquifer in such a case (2.5 yr, Table 8). The total travel time in the unlikely event of release into a rapidly conducting aquifer would be much longer, however. This is because of the time necessary to travel the vertical distance between repository and aquifer horizons. Vertical travel time was ignored in these calculations.

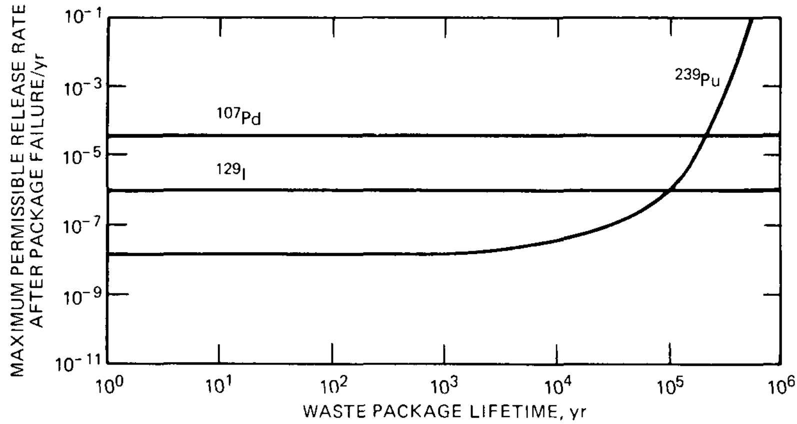


Figure 5 Effects of varying package lifetime. Travel time for all nuclides is 1 or 10^4 yr.

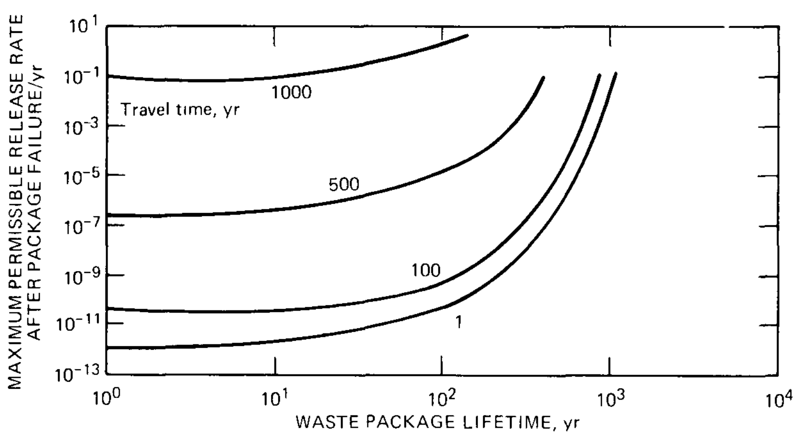


Figure 6 Effects of varying package lifetime and travel time for ^{90}Sr.

Relationship Between Maximum Permissible Release Rate and Waste Package Lifetime. The effect of varying waste package lifetimes (before any leakage occurs) was discussed briefly in the previous section. Figures 5 and 6 were constructed to illustrate further the package lifetime requirement for specific hazardous nuclides identified under Cases 1, 2, and 3. Figure 5 shows the maximum permissible release rate after package failure for ^{129}I, ^{107}Pd, and ^{239}Pu, assuming travel times of either 1 or 10^4 yr and water flow of 3×10^7 l/yr. As can be seen, there is little advantage to be obtained in terms of maximum permissible release rate if the waste package lasts 1000 yr instead of 100 yr. Indeed, significant gains are obtained only if the package lasts several times the half-life of the migrating nuclide. For ^{129}I ($t_{1/2} = 1.7 \times 10^7$ yr), and ^{107}Pd ($t_{1/2} = 7 \times 10^6$ yr), it seems very unlikely that waste package performance could be assured for sufficient time. For ^{239}Pu, significant advantage could be obtained with a 100,000-yr waste package for the short travel times used in the calculation of Fig. 5. No advantage would accrue, however, from a 1000-yr as opposed to a 100-yr package.

It is apparent from Fig. 5 that a 1000-yr waste package would have significant impact only on the maximum permissible release rates of those nuclides that have half-lives substantially less than 1000 yr. As an illustration of this observation, Fig. 6 was constructed for ^{90}Sr, one of the most important short-lived fission products in spent fuel.

Curves of the maximum permissible release rate of ^{90}Sr (after package failure) as a function of waste package lifetime for different values of travel time are shown in Fig. 6. It can be seen that for travel times less than 1000 yr, there is considerable advantage to be obtained from a 1000-yr as opposed to 100-yr waste package lifetime. Strontium-90 did not, however, appear to constitute an important hazard in Cases 1, 2, and 3 because of its long travel times, even in the aquifer-well, Case 3. If, for example, water is flowing at the extremely high velocity of 1000 m/yr, ^{90}Sr would still take on the order of 1000 yr to travel 650 m in the unconfined aquifer. Thus the strong sorption of ^{90}Sr on basalt and other silicate rocks means that it is extremely difficult to envisage a scenario in which ^{90}Sr reaches the biosphere before it decays to very low levels of activity. If engineering was required for the unlikely event of 1- or 100-yr travel times, however, then 1000-yr containment would greatly increase maximum permissible release rates after package failure.

In conclusion, it appears that, for nuclides identified under Cases 1, 2, and 3 as constituting the greatest long-term hazards to life, there is little advantage to be obtained from a 1000-yr as opposed to 100-yr waste package. The principal effect is not on peak discharge rate, but on time of peak discharge. Engineering the package to last 1000 instead of 100 yr postpones the time of peak discharge by 900 yr without greatly altering the height of the peak.

Influence of Radionuclide Travel Time on Maximum Permissible Release Rate. The time taken by nuclides released in the repository to reach the biosphere can be altered in several different ways. The characteristic travel time would, for example, be increased by increasing the distance to the biosphere, by increasing the degree of sorption (K_d) on the host rock, or by decreasing the hydraulic conductivity of the host medium. The incentives for increasing travel times are obvious for many nuclides. If, for example, the travel time of ^{90}Sr is greater than 1000 yr, equivalent to a few tens or hundreds of meters of path length, then the nuclide completely decays before it reaches the biosphere. This kind of observation is, of course, the prime incentive for disposal in deep mined repositories. For many of the nuclides that constitute long-term hazards, however, travel time has to become very long before significant advantage accrues from radioactive decay. A plot of maximum permissible release rate vs. travel time after package failure (Fig. 7) indicates a result analogous to that obtained from Figs. 5 and 6. The travel time for the nuclide of concern has to become much longer than its half-life before the maximum permissible release rate starts to increase significantly. For ^{239}Pu a travel time of 6×10^5 years would result in a maximum permissible release rate in the repository of $\geqslant 10^{-1}$/yr. For Cases 1 and 2, a travel time of 6×10^5 yr for plutonium ($K_d = 50$) would correspond to distances of 48 m and 4.8 km, respectively. Thus, if flow were restricted to massive basalt units, ^{239}Pu would essentially be fixed within

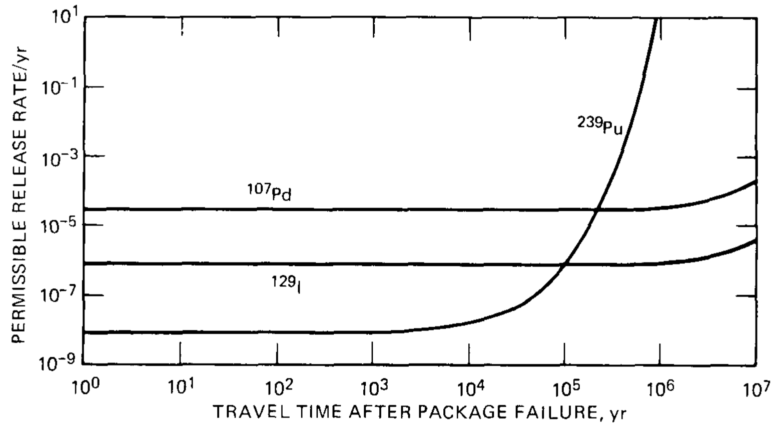


Figure 7 Effects of varying travel time.

the repository until it had all decayed. Leakage into an interflow zone (Case 2) could, however, result in considerable migration of the ^{239}Pu at hazardous levels unless the engineered barrier system confined it at release rates on the order of $\leqslant 10^{-7}$/yr (Fig. 7).

For ^{129}I, which as I^- has very little tendency to sorb on silicate surfaces, travel times much longer than 10^7 yr are required to affect significantly the maximum permissible release rate (Fig. 7) calculated assuming a water flow of 3×10^7 l/yr as before. Thus for ^{129}I the release requirements on the waste package are extremely stringent and little benefit is likely to be derived from increasing the path length to the biosphere because of the difficulty of obtaining the necessary long travel times. The release requirements of about 10^{-6}/yr (given the water flow described) will, therefore, hold until all the ^{129}I has been released.

Solubility Constraints on Radionuclide Release Rates.

The maximum permissible release rates of radionuclides calculated for Cases 1, 2, and 3 may be fairly readily converted to maximum permissible concentrations within the waste repository. This is done by converting release rate in curies per year into grams per year and dividing by the water flow. There are several ways in which the waste package may be used to achieve the desired performance level. One way would be to produce a canister or overpack with a very long and predictable life in a repository located in basalt. Another would be to use a waste form with accurately calibrated and acceptable leaching behavior. A third would be to make the backfill reactive with key radionuclides, and a fourth would make the backfill of extremely low permeability to groundwater. In this context it is pertinent to ask, Which are the key radionuclides for which the engineered barrier system must be most effective? This question was answered in part by Barney and Wood (1980). In this study, nuclides with the lowest maximum permissible release rates were identified. Succeeding studies must now consider how the waste package might be engineered to keep release rates of key radionuclides below maximum permissible levels. This is not particularly important, however, for nuclides for which the maximum permissible concentration in the repository is well above the solubility of the most stable compound

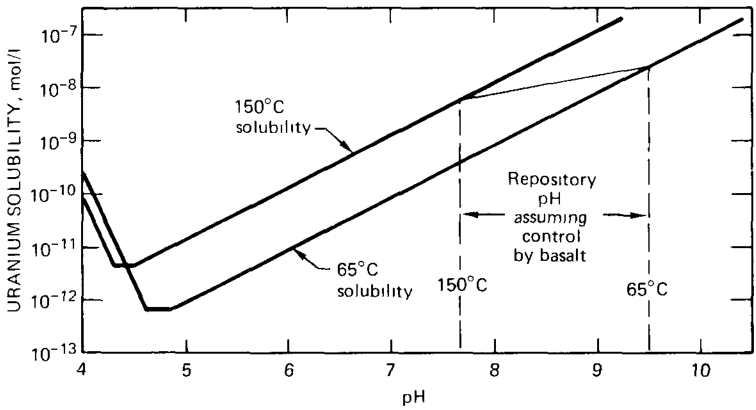


Figure 8 UO_2 solubility in Hanford groundwater.

of the element under repository conditions. Assuming equilibrium, nuclides will be controlled at concentrations below the maximum permissible level by direct precipitation. In such cases (Fig. 2, step 6), the engineered barrier system will not need to perform any specific function.

Solubility and thermodynamic data are available for some but not all of the radionuclides of concern in a repository located in basalt. To illustrate the possible influence of solubility constraints, let us consider ^{238}U, an isotope that, together with its daughters ^{234}U, ^{226}Ra, and ^{210}Pb, has been found to constitute one of the more important hazards in a repository in basalt. Langmuir (1978) produced a compilation of thermodynamic data on solid and aqueous uranium species which may be used to calculate the solubility of uranium in groundwater over a wide range of conditions of temperature and oxygen partial pressure. Hodges (in Smith et al., 1980) showed that, some time after repository closure, conditions will be extremely reducing, with oxygen partial pressures controlled close to those of the Ni-NiO oxygen buffer. Under such conditions, the stable uranium solid will be either UO_2 (silica-free groundwater) or $USiO_4$, and the dominant aqueous species (calculated from Langmuir, 1980) will be U(IV) complexes. The solubility of UO_2 at 65 and 150°C is shown in Fig. 8 for groundwater from a potential repository unit (Wood, in Smith et al., 1980). Uranium oxide was used for the calculations because it is the most important component of spent fuel and will therefore, be present within the repository. If equilibrium pertains, the uranium solubility should be lower because groundwaters are rich in silica and should precipitate coffinite ($USiO_4$). Figure 8 thus represents conservative solubility estimates. Hodges (in Smith et al., 1980) estimated that repository pH could, under conditions of rapid flow, be depressed to about 4. The more likely pH, however, is close to that obtained under steady-state conditions in recirculating hydrothermal experiments (Hodges, in Smith et al., 1980). These pH's, shown in Fig. 8, yield a maximum uranium solubility of about 3×10^{-8} mol/l or 7 ppb. This would be the maximum amount of uranium present in solution within the repository.

If we make the conservative assumption that every liter of water passing through the repository is saturated in uranium, about 6×10^{15} l would be required to dissolve all the uranium present in the inventory of Table 3. This leads to the following maximum leach rates of ^{238}U and ^{234}U in the three cases considered earlier:

	Maximum possible uranium leach rate	Maximum permissible uranium leach rate
Case 1	5×10^{-12}/yr	1×10^{-7}/yr
Case 2	5×10^{-9}/yr	4×10^{-5}/yr
Case 3	5×10^{-7}/yr	$>10^{-3}$/yr

Thus simple solubility constraints should lead to release rates of uranium which are several orders of magnitude below the maximum permissible release rate. Preliminary calculations indicate that similar conclusions may also apply to Pu, Am, ^{79}Se, and ^{94}Zr, but not to ^{129}I. It is apparent that generating data on solubility and related thermodynamic data will lead to much clearer definitions of the hazardous nuclides in a repository in basalt. Solubility constraints for other geologic repositories will be rather different from those obtained for basalt. Rocks such as granite and salt (dominantly NaCl) have a much lower capacity to reduce uranium to U(IV) than does basalt. If uranium becomes oxidized to U(VI), its solubility increases dramatically (Rich, Holland, and Petersen, 1977), and its release rate cannot be controlled by the solubility mechanism.

The performance requirements for different components of the engineered barrier system will obviously be much more clearly defined when diagrams similar to Fig. 8 can be constructed for all important radionuclides in the waste inventory.

Conclusions

Solutions to the one-dimensional transport equation provide a rapid means of estimating conservative performance requirements for the engineered barrier system. Three potential release and transport scenarios have been modeled in this way to determine the most hazardous nuclides in the waste inventory and to estimate maximum permissible release rates for the waste package. The main criterion used to define maximum permissible release rate was that solutions discharging to the accessible environment should contain less than the RCG (Nuclear Regulatory Commission, 1976) of all nuclides. The three scenarios and most stringent release rates were:

1. Transport through 25 km of basalt to the discharge point; lowest permissible release rate = 2×10^{-9}/yr for ^{129}I
2. Transport through 25 km of interflow zone to the discharge point (modeled as if the repository were actually built in the interflow zone); lowest permissible release rate = 2×10^{-7}/yr for ^{237}Np
3. Transport through the unconfined aquifer to a discharge point 1750 m from the repository center (modeled as if the repository were built in the aquifer); lowest permissible release rate = 1×10^{-6}/yr for ^{241}Am

The estimates are conservative by at least one to two orders of magnitude because of the effects of lateral dispersion. In the first case the volume of discharging solution (3×10^4 l/yr) is too small to provide a significant water resource, but the latter two examples could be important in this context. Thus release rates on the order of those calculated under the second and third scenarios should provide a design basis for the engineered barrier system.

For the important nuclides identified under Cases 1, 2, and 3, there is little advantage to be gained by producing a 1000-yr rather than a 100-yr waste package. Waste package lifetime has a significant effect on the maximum permissible release rate of a nuclide only if the lifetime is several times the half-life of the nuclide. A 1000-yr waste package is important only for short-lived nuclides, such as ^{90}Sr ($t_{1/2}$ = 28 yr), and then only if the nuclide travel time to the biosphere is ignored. None of the short-lived fission products, such as ^{90}Sr, appeared as hazards because their travel times to the accessible environment are all much longer than their half-lives. Thus, even if they escape the waste package instantaneously, they are unlikely to constitute important hazards because they decay before reaching the environment.

For a repository in basalt, it appears that the discharge rates of uranium and its daughters (e.g., ^{226}Ra) will be kept well below the RCG because of the low solubility of UO_2 under reducing conditions. Thus it is unlikely that the barrier system will have to immobilize chemically the uranium in spent fuel. Similar conclusions may well be reached for other actinides and fission products when adequate solubility data are available.

The barrier system must operate primarily to immobilize nuclides such as ^{129}I, for which there appear to be almost no constraints on concentration in solution. The performance requirements for nuclides such as ^{129}I should become considerably less stringent, however, when more quantitative transport models can be constructed for specific sites.

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Borosilicate Glass as a Matrix for the Immobilization of Savannah River Plant Waste

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The reference waste form for immobilization of Savannah River Plant (SRP) waste is borosilicate glass. In the reference process, waste is mixed with glass-forming chemicals and melted in a Joule-heated ceramic melter at 1150°C. Waste glass made with actual or simulated waste on a small scale and glass made with simulated waste on a large scale confirm that the current reference process and glass-former composition are able to accommodate all SRP waste compositions and can produce a glass with:

- **High waste loading**
- **Low leach rates**
- **Good thermal stability**
- **High resistance to radiation effects**
- **Good impact resistance**

Borosilicate glass has been studied as a matrix for the immobilization of SRP waste since 1974. This paper reviews the results of extensive characterization and performance testing of the glass product. These results show that borosilicate glass is a very suitable matrix for the immobilization of SRP waste.

Introduction

Radioactive waste produced from reprocessing nuclear fuel for defense programs at Savannah River Plant (SRP) is stored in large underground tanks on the plant site. This alkaline waste is made up of three parts. The bulk of the waste actinides and fission products is contained in an insoluble sludge of hydroxides and hydrous oxides of iron, aluminum, and manganese (Wallace, Hull, and Bradley, 1973; Stone, Kelley, and McMillan, 1976; Stone, 1976). The rest of the waste is in the form of either a crystalline salt cake or a nearly saturated supernatant salt solution. The supernatant solution contains nearly all the radiocesium and traces of other radionuclides (Ondrejcin, 1974).

Methods for immobilizing SRP waste for long-term storage are being developed at the Savannah River Laboratory (SRL). Waste immobilization is currently envisaged in two stages to reduce the initial capital investment and allow the most efficient use of limited resources. Since approximately 95% of the long-lived (>100-yr half-life) radionuclides are contained in the insoluble sludge, the first stage will be designed to immobilize the sludge. In the second stage, the radionuclides removed from high-level waste salt will be immobilized beginning a few years later.

According to the current reference process, sludge (washed essentially free of soluble salts) is mixed with glass-forming chemicals in the form of premelted glass frit, and the slurry is heated to drive off excess water and then fed to a Joule-heated ceramic melter. Here the slurry dries and forms a glass, which is continuously poured out of the melter into stainless steel canisters. The canisters are decontaminated, welded closed, and stored on an interim basis onsite. Eventually, they will be shipped to a federal repository for long-term storage.

Since 1977, SRL has been performing continuous melting tests using simulated waste on both large- and small-scale equipment. In early 1979 SRL began processing actual waste in a small-scale glass plant in hot cells (capacity approximately 0.45 kg or 1 lb/hr).

The experience gained in these tests, coupled with an extensive body of laboratory research on glass quality, showed that:

- The process and the glass composition can accommodate all SRP waste compositions.
- The glass produced has a high waste loading.
- The waste glass is durable; i.e., it has low leach rates.
- The waste glass has good thermal stability; no significant devitrification will occur during normal processing or storage.
- The waste glass resists impact, producing only a very small quantity of respirable fines.
- The waste glass is virtually unaffected by radiation.

Glass Composition

The average composition of the glass waste form is shown in Table 1. According to the reference process, a slurry containing 40 wt.% solids (35 wt.% waste and 65 wt.% glass-forming chemicals in the form of premelted glass frit) is fed directly to a continuous glass melter. The slurry dries and is melted at 1150°C to produce a borosilicate glass containing 29.0 wt.% waste oxides and 252 kCi/m^3 (producing 0.747 kW/m^3 of glass). The glass is poured into steel canisters and allowed to cool.

The present reference glass frit composition, Frit 131 (Table 2), was developed on the basis of leach rate, tendency to devitrify, insensitivity to poor mixing (waste solubility) and waste composition, and viscosity (Plodinec, 1980). Because of the variety of chemical separation processes run since the early 1950s at SRP, the composition of the waste can vary widely (Table 3). The ability of borosilicate glass to accommodate all the variations was an important factor in its selection as the reference waste form.

Currently, SRL is working with the commercial glass industry to develop higher silica glass frit compositions.

Table 1 Composition of SRP Waste Glass

Component	Source*	Amount, wt.%
SiO_2	F + W	41.1
Fe_2O_3	W	14.5
Na_2O	F + W	13.0
B_2O_3	F	10.4
Li_2O	F	4.0
MnO_2	W	4.0
Other solids†	W	3.0
Al_2O_3	W	2.8
NiO	W	1.8
MgO	F	1.4
U_3O_8	W	1.4
CaO	W	1.1
TiO_2	F	0.7
ZrO_2	F	0.4
La_2O_3	F	0.4

**F is Frit 131, and W is average waste.*
†"Other solids" include undissolved salts and fission products.

Table 2 Composition of Frit 131

Component	Amount, wt.%
SiO_2	57.9
Na_2O	17.7
B_2O_3	14.7
Li_2O	5.7
MgO	2.0
TiO_2	1.0
La_2O_3	0.5
ZrO_2	0.5

Table 3 Composition of SRP Waste

Component	Amount, wt.% Average	Possible range
Fe_2O_3	48	15-56
MnO_2	12	4-15
Other solids*	11	0-12
Al_2O_3	9	1-51
NiO	6	2-10
U_3O_8	5	0-13
CaO	4	1-6
SiO_2	3	0-10
Na_2O	2	1-7

*"Other solids" include fission products and undissolved salts.

Although it is difficult to make absolute statements at this stage, it appears that the leach rate can be reduced by about an order of magnitude. These compositions can be processed at 1150°C, and, because of their lower alkali content and higher viscosities, they may also offer lower melter corrosion rates and reduced volatilities.

Glass Leaching

The rate of release of radionuclides from the glass waste form is the product of the leachability [g/(m^2 · d)] of the glass and the surface area (m^2) of the glass waste form. Experiments at SRL have shown that the leachability of the glass depends on the pH of the leachant, leaching temperature, and extent of devitrification. Under expected repository conditions, both short- and long-term experiments indicate that leach rates should be low. Experiments at SRL have also shown that pressure and radiolysis effects on the leachability are not significant. The surface area of the glass form is determined by the cooling rate of the glass and the interactions between the glass and canister. Experiments at SRL indicate that the surface area of the glass in its canister should be no more than five times the geometric surface area.

Leachability. The pH of the leachant is probably the single most important variable in determining the leachability. As Fig. 1 shows, at room temperature glass leaching is at a minimum in the pH range of 4 to 10. Within this pH range, the primary leaching mechanism is probably ion exchange between the leachant and the glass; whereas, outside this pH range, another leaching mechanism, network dissolution, becomes predominant. As Fig. 1 shows, this minimum leaching pH range coincides well with the pH of most repository groundwater compositions (Wicks, 1981). In 28-d standard (MCC-1) leach tests at 40°C (slightly above ambient temperature in a salt or granite repository), leachabilities range from 0.01 to 0.1 g/(m^2 · d) and average about 0.03 g/(m^2 · d).

Increasing the temperature to 90°C (which represents the maximum temperature at glass-water contact in a granite or salt repository) narrows this optimum range to pH 5 to 9. It also increases the minimum leachability about five times. Even at the peak temperature of 90°C (reached 25 yr after emplacement), however, the leachability is still low. In 28-d tests at 90°C, leachabilities range from 0.05 to 1.0 g/(m^2 · d) and average about 0.3 g/(m^2 · d).

If the glass is deliberately devitrified, it can lead to an increase in leachability of approximately 10 times. At 90°C for extensively devitrified samples, leach rates average 3 g/(m^2 · d) (Robnett and Wicks, in preparation). In general, this increase in leachability occurs only if the devitrified phases are silicates. However, large-scale tests showed that, for well-mixed feeds (as in the reference process), none of these phases (or any other) were formed in full-size canisters allowed to cool by natural

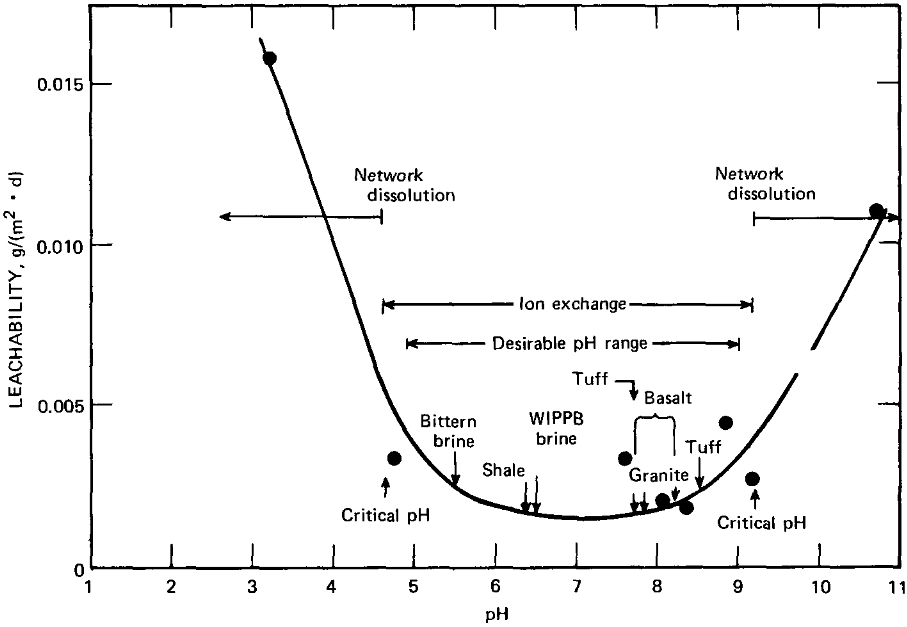

Figure 1 Effect of pH on glass leachability, with leachability based on silicon. The 5-d static tests used TDS/131 waste glass at 23°C. Glass is Frit 131 plus "average" waste, and area-to-volume ratio is 10/cm. The "desirable pH range" indicated is for groundwaters during storage.

convection (Plodinec, 1980). Thus under normal processing conditions devitrification will not occur. Also, because of the low thermal power content of these glasses, devitrification will not occur during long-term storage.

Recent experiments have shown that pressure does not significantly affect the leachability of waste glass (Wicks et al., in preparation). In fact, as shown in Table 4, the leachability actually decreased approximately two times at higher pressure. This is important because the glass waste form could be exposed to elevated pressures (either lithostatic or hydrostatic) in the repository environment.

It has also been shown that radiolysis of the leachant does not significantly affect the leachability. The radiolysis tests were done under a variety of conditions (Walker et al., 1981) (e.g., variations in pH, time, temperature, surface-area-to-volume

Table 4 Effect of Pressure on Leachability*

	Leachability, $g/(m^2 \cdot d)$	
Pressure, atm	**Silicon**	**Sodium**
1.0	0.75	1.26
	0.77	1.30
102.0	0.45	0.68
	0.44	0.63

**Tests run 30 d at 90°C. Ratio of surface area to leachant volume is 0.1/cm. Leachant is ultra-pure water.*

Table 5 Glass Leach Rates in Various Media*

Element	Leach rate, $g/(m^2 \cdot d)$		
	Deionized water	Silicate water	Brine
Si	0.55	0.21	0.21
B	0.79	0.38	0.31
Na	0.89	1.08	†
Cs	1.39	0.83	0.24
Fe	0.007	0.05	0.01
Sr‡	<0.4	<0.4	<0.4
Mass loss	0.64	0.34	†

MCC-1 leach tests; 28 d at 90°C.
†Interference from brine in final analysis.
‡None detected in leach solutions. Maximum leachability calculated on the basis of detection limits for strontium.

ratio, alpha, beta, and gamma radiation, and dose rates), and all lead to the conclusion that radiolysis of the leachant causes no significant changes in the leaching behavior.

Dran, Maurette, and Petit (1980) reported that external irradiation of soda-lime silica glasses by lead ions (simulating alpha-recoil damage) to doses greater than 10^{12} particles/g increased the degree of penetration of the glass by simulated brine solutions. They postulated that this would mean a great increase in the release rate of radionuclides into solution; i.e., the leachability of borosilicate glass would be greatly increased. Even though defense waste glasses will not encounter the dose at which this effect was observed, SRL is investigating this effect. No increase in leachability has been observed in lead or xenon ion-irradiated glasses in experiments conducted to date, nor in ^{244}Cm-doped glasses at comparable doses (Bibler and McDonell, in preparation).

Longer term tests have also been performed in a variety of media, both at 90°C, the maximum leaching temperature in a repository, and at 25°C, the long-term condition in the repository (Dukes et al., 1980; Walker et al., 1980). At 90°C, leachabilities are approximately 0.3 $g/(m^2 \cdot d)$ after 28 d (Table 5). At 25°C, leach rates of actual waste glasses are approximately 10^{-3} $g/(m^2 \cdot d)$ (Table 6). In both cases the leach rate decreases with time, as shown in Table 6.

Surface Area. The fracturing of large glass waste forms during cooling in the production increases the surface area available for leaching. Acoustic emission and visual analyses showed that cracking in the bulk glass occurs in the annealing range (450 to 550°C). Surface cracking occurs at lower temperatures because of interactions between the glass and the canister (Smith and Wiley, 1980). Experiments have shown that bulk cracking can be minimized by cooling the glass form slowly through the annealing range. Surface cracking can be minimized by:

- Matching coefficients of expansion of the glass and the canister (e.g., using carbon steel, titanium, or ceramic as primary canisters or as liners for canisters)
- Using crushable liners (e.g., aluminosilicate papers)
- Using liners that allow the glass to slip along the canister wall (e.g., graphite)

Laboratory tests have shown that cracking can be easily controlled so that surface area increases only two to five times (in comparison with a monolith).

Thermal Stability of Glass

Thermal stability of the waste form involves both:

- The ability of the waste form to resist slow processes associated with its own self-heating in a repository
- The ability of the waste form to resist rapid changes associated with high-temperature events, such as fires

Table 6 Long-Term Leaching Experiments*

	Leachability,† g/(m^2 · d)		
Time, d	Plutonium	Strontium	Cesium
4	2.3×10^{-3}	4.5×10^{-3}	1.5×10^{-3}
28	0.9×10^{-3}	2.5×10^{-3}	3.2×10^{-3}
100	0.9×10^{-3}	2.4×10^{-3}	2.3×10^{-3}
500	0.7×10^{-3}	2.2×10^{-3}	2.9×10^{-3}
1000	0.5×10^{-3}	1.7×10^{-3}	2.2×10^{-3}

**Room temperature; area-to-volume ratio approximately 0.1/cm.*

†Average for two waste types and three leach waters (pH 7, nine buffers, distilled water).

Calculations have shown that the heat-generation rate of SRL waste glass is too small to affect product quality. Experimental data indicate that only severe high-temperature events can affect the product, primarily by devitrification.

Self-Heating Effects. Members of the Reference Repository Conditions Interface Working Group, set up by the Office of Nuclear Waste Isolation, calculated that SRL waste emplaced in a granite or salt repository would reach a maximum surface temperature of about 90°C 25 yr after emplacement and then slowly decay (Fig. 2). After 100 yr, the glass temperature would be 70°C. At these temperatures, the rates of processes such as devitrification are insignificant (Boulos et al., 1980). Since these calculations were performed for potential granite, salt, and tuff repositories, self-heating should have no effect on waste glass in any repository environment now under consideration for defense waste.

Effects of High-Temperature Events. High-temperature events might cause concern about glass volatility, expansion, and devitrification. Only devitrification can significantly affect the glass, however, and that would occur only in a rather severe high-temperature event.

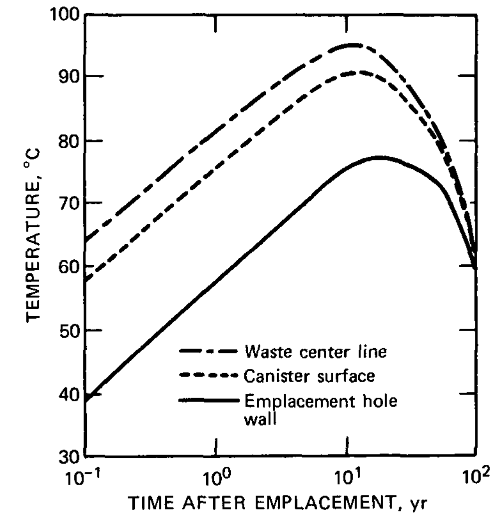

Figure 2 Very near-field thermal response for a stage 1 defense high-level waste repository with an areal thermal loading of 8.4 W/m^2 in granite.

Volatility depends on temperature, atmosphere above the glass, and the composition of the glass. Savannah River Laboratory has not performed a comprehensive study at intermediate temperatures (<900°C), but at 900°C only 19 mg/hr of volatiles would be released from a canister of glass (Wilds, 1978). Since volatility varies exponentially with temperature, at lower temperatures the amount released would be negligible. Simi-

Table 7 Crystalline Phases Produced by Heat Treatments

Temperature, °C	Crystalline phases formed
Samples Heated 1 d	
500	None
550	19% acmite
600	46% acmite
650	48% acmite
800	38% acmite, 14% spinel
900	27% spinel
Samples Heated 1 Week	
500	None
550	54% acmite, Li_2SiO_3
600	44% acmite, 6% Li_2SiO_3

larly, volume changes associated with expansion (<1%) would be so small that any stress generated in the canister would be negligible.

In recent experiments at SRL, devitrification was investigated by heating various waste-glass compositions for 1 d at temperatures from 500 to 900°C. Crystalline phases were identified by X-ray diffraction and scanning electron microscope microprobe, and the amount was quantified by comparing sample X-ray peak heights with those of standards (Table 7). Acmite ($NaFeSiO_4$) and spinels (e.g., $NiFe_2O_4$) were the main crystalline products formed in this temperature range. At 500°C, no crystalline material was found for any of the compositions studied, even after a 1-week heating period. Samples were also heated for 1 week at 550 and 600°C to examine devitrification in more detail. Under these conditions, a small amount of Li_2SiO_3 also formed. In a waste-glass composition high in aluminum, a small quantity of nepheline ($NaAlSiO_4$) was also observed. The formation of the silicate phases increased the leachability (as evidenced by Na, Si, Cs, and Sr), but only to about 2 to 4 g/(m^2 · d). The fact that crystallinity did not appear in the samples held at 500°C indicates that severe high-temperature events are necessary to cause devitrification (Robnett and Wicks, in preparation).

Mechanical Stability of Glass

J. L. Jardine, Argonne National Laboratory, who recently measured the impact resistance of SRL waste glass, found that, for an impact energy density of 10 J/cm^3, 0.14 wt.% respirable fines (<10 μm diameter) were generated. This compares favorably with values for ceramic waste forms, such as SYNROC, which produced 0.16 wt.% respirable fines for the same impact energy density.

Radiation Stability of Glass

During long-term storage (10^6 yr), the glass waste form will receive a dose of 5×10^{10} rads from beta and gamma radiation and 10^{18} alpha/g of glass. Results of radiolysis studies showed that ^{60}Co gamma and ^{244}Cm alpha radiations have minimal and insignificant effects on the density and leachability of the glass.

Density. A sample of SRL waste glass was doped with 2 wt.% ^{244}Cm. At a total dose of 9.3×10^{17} alpha/g (simulating 9×10^5 yr storage of actual SRP waste glass), the density

Table 8 Effects of Radiation on the Density of Waste Glass

Radiation	Dose	Equivalent age, yr	Result
Gamma	6×10^{10} rads	$>10^6$	No change
Alpha	9.3×10^{17}/g	9×10^5	Density decreased 1.0%

Table 9 Effect of Gamma Radiolysis During Leaching in Deionized Water*

	Leach rate,† $g/(m^2 \cdot d)$	
Constituent	Unirradiated‡	Irradiated‡
Silicon	1.9×10^{-2}	2.3×10^{-2}
Boron	2.4×10^{-2}	2.4×10^{-2}
Lithium	2.3×10^{-2}	2.3×10^{-2}
Sodium	2.0×10^{-2}	1.8×10^{-2}

1.5×10^6 rads/hr, $T = 45°C$.
†Based on aliquots removed during leaching.
‡Based on least-squares slope of line defined by four data points. Mass of the glass, 8.4 g; Sa, 20 cm^2; and leachant volume, 80 ml.

decreased by 1.0% (the glass is expanding). Dose response is similar to that predicted from earlier results (Bibler and Kelley, 1978).

A sample of SRL waste glass was irradiated in the SRL ^{60}Co Irradiation Facility to a dose of 6×10^{10} rads (simulating $>10^6$ yr of storage). There was no change in density. These two density tests are summarized in Table 8.

Leachability. The results of leach tests were mentioned in the section on glass leachability, but a few specific examples should be discussed. Leach rates were determined for samples of SRL waste glass leached at 40°C in deionized water for up to 25 d. One sample was irradiated in the ^{60}Co Irradiation Facility during leaching, and one was not. As Table 9 shows, there was no significant effect of gamma irradiation.

A sample of SRL waste glass doped with 2 wt.% ^{244}Cm was leached for over 100 d at 25°C according to a modified standard leaching procedure specified by the International Standards Organization. The leachability of this high-dose-rate glass is compared with a low-dose-rate glass containing ^{239}Pu in Fig. 3. Although

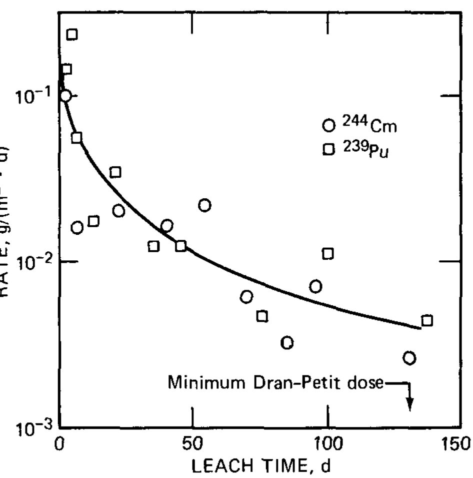

Figure 3 Time dependence of leach rate of ^{244}Cm or ^{239}Pu from SRP waste glass at a temperature of 25°C. (Note that ^{244}Cm glass is 23,000 times more radioactive than ^{239}Pu glass.)

Table 10 Effect of External Xe^{2+} Ion Irradiation on Leachability of SRP Waste Glass*†

Dose,‡ ions/cm^2	Equivalent storage time, yr‡	Leach rate,§ $g/(m^2 \cdot d)$ Based on silicon 0–5 hr	Based on silicon 5–10 hr	Based on sodium 0–5 hr	Based on sodium 5–10 hr
0	0	0.58	0.47	0.97	0.33
3×10^{13}	$>5 \times 10^6$	0.70	0.58	1.4	0.52

Two tests at 90°C in static deionized H_2O for 5 hr; surface-area-to-volume ratio, 0.1/cm.

†Frit 131/TDS-3A waste glass.

‡Dran–Petit threshold dose, approximately 5×10^{12} ions/cm^2.

§Approximately 500 Å leached per test; range of 160 keV Xe^{2+} ion is approximately 1000 Å.

the dose to the Cm-doped glass is 23,000 times greater, the leachabilities of the two glasses are not appreciably different.

The minimum dose for the Dran–Petit effect mentioned in the section on leaching is also indicated in Fig. 3 (Dran, Maurette, Petit, 1980). For glasses with a total dose greater than that indicated, greatly enhanced leaching (20 times as great) was predicted. As Fig. 3 shows, this is not the case.

To further examine this potential effect, we irradiated SRL waste-glass samples with xenon or lead ion beams to very high doses. (Dran, Maurette, and Petit used lead ions; xenon ion irradiations were performed to look for possible effects specific to lead.) Some preliminary data are shown in Table 10. In these tests the glasses were leached for two 5-hr periods, and the leachant was analyzed after each period. In 10 hr, the observed leach rate corresponded to complete penetration of the irradiated surface layer by water. In other words, any effect would be most accentuated in these tests. As Table 10 shows, the glass with the highest dose (approximately 6 times the minimum dose) has a leach rate only about 1.4 times that of the unirradiated sample. Since this sample received a dose equivalent to $>5 \times 10^6$ yr of storage time for SRL waste glass, it appears that the effect observed by Dran, Maurette, and Petit may not be important for the long-term storage of SRP waste glass—at least for repositories other than salt. Tests are being conducted at several temperatures, and especially in brine, to completely evaluate the importance of the effect.

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Sandia Studies of High-Level Waste Canisters and Overpacks Applicable for a Salt Repository

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An experimental program to develop candidate materials for use as high-level waste (HLW) overpacks or canisters in a salt repository has been in progress at Sandia National Laboratories since 1976. The main objective of this program is to provide a waste package barrier having a long lifetime in the chemical and physical environment of a repository. This paper summarizes the recent corrosion and metallurgical study results for the prime overpack material, TiCode-12, in the areas of uniform corrosion (extremely low rate and extent); local attack, e.g., pits and crevices (none were found); stress corrosion cracking susceptibility (no significant changes in macroscopic tensile properties were detected); hydrogen sorption–embrittlement effects; effects of gamma irradiation in solution; and sensitization effects (testing is still in process in the last three areas). Previous candidate screening analyses on other alloys and recent work on alternate overpack alloys are reviewed. All phases of these interrelated laboratory, hot-cell, and field experimental studies are described.

Introduction

A program to investigate the corrosion and metallurgical behavior of candidate alloys for use as high-level waste (HLW) canisters or overpacks is being conducted at Sandia National Laboratories. Materials selected must maintain their barrier integrity for design lifetimes of several hundred to a thousand years or more. A specific purpose of our studies was to evaluate potential metallic waste package barriers under the environmental conditions expected in a bedded or domed salt repository. Environmental conditions applicable for waste containers placed in subseabed sediments are also included. The primary emphasis in this paper is on summarizing the results applicable to salt repositories; data for materials subjected to subseabed conditions are also presented, however.

Objectives of this program are (1) to provide materials research and development results in support of waste package design efforts and (2) to identify the kinetics and mechanisms of canister/overpack degradation so that performance can be modeled and credible predictions of lifetimes can be made. A limited number of selected candidate alloys are being tested and developed in detail so that prototypical waste packages can be fabricated and tested in situ.

The waste canister is a physical barrier for containment. It must provide sufficient mechanical strength for transportation and handling operations during both disposal and retrieval, should retrieval be required. It must be able to survive the high-temperature environment of waste processing, e.g., glass melting by the in-can melt or ceramic heater processes. The canister, or the canister in conjunction with the overpack, must be designed to survive for long periods of time under the environmental conditions of repository isolation. The canister material of choice at the present time is stainless steel 304L; it meets the requirements of the in-can melt process. This material is not expected to have a significantly long lifetime in a salt repository because of chloride-induced stress corrosion cracking. Mild steel could be substituted if the in-can melt process was not used.

The overpack is another physical barrier, directly encapsulating the waste canister (like a second canister). Its purpose is to enhance corrosion resistance and durability, thus providing barrier effectiveness for a long time period. The overpack may be much thinner than the canister.

The corrosion and metallurgical studies at Sandia National Laboratories have been oriented primarily to evaluating the performance of overpack materials. We have assumed that the canister material will be selected by the waste processor. Corrosion results described here are applicable to either metallic barrier.

The experimental studies and results described for both salt and subseabed environments have been in progress at Sandia since 1976. The salt studies were initially specific to the Waste Isolation Pilot Plant, a proposed defense waste research and development facility in deep underground beds of rock salt, located in southeastern New Mexico. In 1980, however, the experimental canister-overpack studies were broadened and made applicable for a generic bedded or domed salt repository. These studies are supported by and are part of the National Waste Terminal Storage (NWTS) program.

Table 1 Experimental Program Phases

Candidate alloy screening tests
Initial screening
Design alternates
General corrosion
Uniform and localized corrosion
Corrosion mechanisms
Radiation effects
Environmental cracking and embrittlement
Physical and mechanical metallurgy
Laboratory studies
Overpack thickness and installation
Advanced hot-cell and field testing

The overall canister–overpack experimental program consists of multiple phases conducted in parallel (listed in Table 1). Experimental descriptions (i.e., materials and environment, apparatus, and specimen preparation) are summarized elsewhere (Braithwaite, Magnani, and Munford, 1980). In the remainder of this paper, we briefly summarize data reported previously (Braithwaite, Magnani, and Munford, 1980; Braithwaite and Molecke, 1980), review recent experimental results, and describe the programmatic status of all phases of this study.

Candidate Alloy Screening Results

The applicable chemical and physical environment (temperature, pressure, radiation, solution composition(s), etc.) under which an HLW package must maintain its barrier integrity in a salt waste repository has been summarized previously (Braithwaite and Molecke, 1980; Claiborne, Rickertsen, and Graham, 1980). The compositions of brines postulated for salt waste repositories or the subseabed are summarized in Table 2. Brine A is representative of brines intruding into a repository in bedded salt; brine B is more appropriate for domed salt. The most credible condition for a waste package is to be

Table 2 Representative Solution Compositions

	Major constituents, molarity							
	Na^+	K^+	Mg^{2+}	Ca^{2+}	Cl^-	SO_4^{2-}	HCO_3^-	BO_3^{3-}
Brine A	1.8	0.77	1.4	0.02	5.4	0.04	0.01	0.02
Brine B	5.0			0.02	4.9	0.04		
Seawater	0.46	0.01	0.05	0.01	0.54	0.01		

Table 3 Corrosion Rates of Candidate Alloys*

Alloy	Seawater, mm/yr	Brine B, mm/yr	Brine A, mm/yr	Brine A (O_2 = 600 ppm), mm/yr
1018 mild steel	0.4	0.07	1.7	7.0
1018 mild steel (70°C)	†	0.036	0.07	†
1018 mild steel (25°C)	†	0.03	†	†
Corten A steel	0.2	0.05	0.9	†
2¼ Cr-1 Mo steel	0.2	0.1‡	1.0‡	†
Lead	0.3	0.3	0.5	†
Copper	0.07	0.05	0.07	1.2
90-10 Cupronickel§	0.07	†	0.14	0.4
Stainless steel 304L	0.006	0.01	0.018	†
Stainless steel 316L	0.005	†	0.015	†
Stainless steel Nitronic 50	0.003	†	0.008	†
Stainless steel Ebrite 26-1	0.005¶	†	0.016¶	0.24
Monel 400	0.1¶	†	0.03¶	†
Incoloy 825	0.004	†	0.006	†
Inconel 600	0.005	0.007	0.009	†
Inconel 625	0.012**	0.001	0.005	†
Hastelloy C-276	0.0015	†	0.007	0.06‡**
Zircaloy-2	0.001	†	†	†
Ti-50A	0.012	†	0.014	†
TiCode-12	0.001	†	0.003	0.0004

**Temperature was 250°C; pressure, 5 MPa; O_2, ~30 ppb; and time, 28 d.*
†Not tested.
‡Crevice corrosion.
§Poorly adherent, heavy scale formed in both dry and moist salt environment.
¶Pitted after 6 months.
***Pitting corrosion.*

in contact with dry or slightly moist salt for at least the first 1000 yr of its emplacement. The probability that brine will eventually intrude and then saturate the waste package cannot be eliminated, however. Waste containers emplaced in the subseabed would, of course, be in contact with saturated sediments. The vast majority of corrosion tests conducted used brine-inundated conditions as a limiting boundary to the corrosive environment; a small number of tests have used dry or moist salt conditions (Braithwaite and Molecke, 1980).

Initial Screening. The results of our initial screening studies on about 20 candidate alloys were presented earlier (Braithwaite and Molecke, 1980) and are summarized in Table 3.

Table 4 Effect of Temperature on Solution pH*

Temperature, °C	Brine A	Brine B	Seawater
25	6.5	7.2	8.1
100	7.0	7.6	8.6
150	6.3		5.9
200		6.6	5.5
250	3.4	6.4	3.9
270		6.3	3.3

**The pH was measured at 25°C.*

Overall corrosion rates (after 28 d or 6 months of testing) were measured by weight change. Observations of localized attack, i.e., crevice corrosion and pitting, are noted. Most of the accelerated testing was done at 250°C, the approximate maximum expected temperature on the outer surface of the overpack for commercially reprocessed HLW emplaced in a dry salt repository. For most alloys this environment is an overtest condition, as is the high oxygen concentration shown in Table 3 (last column). A maximum temperature of about 150°C is more credible for a repository assumed to be inundated with brine.

In most cases hot brine A was more corrosive than hot seawater, which was, in turn, more corrosive than hot brine B.

Increases in temperature from 25 to 270°C decrease the solution pH, primarily because of the precipitation of magnesium oxysulfates and the concommitant formation of HCl. The effect of temperature on measured pH values is shown in Table 4. In many cases the decrease in pH results in an increase in corrosion rate.

The economic impact of fabricating an overpack out of several of the alloys listed in Table 3 was also calculated and presented elsewhere (Braithwaite and Molecke, 1980). Analyses included materials cost per overpack (assumed to be 0.62 cm thick), material requirements vs. resource reserves, and relative alloy cost per desired lifetime (1, 25, and 300 yr). Some alloys, e.g., Zircaloy, were eliminated from future consideration on the basis of economics alone.

One alloy, 90-10 Cupronickel, which was initially selected as a backup material, was subsequently dropped from consideration when further testing revealed the formation of a poorly adherent, heavy oxide scale in both dry and moist crushed salt environments.

On the basis of corrosion testing and economic considerations, a titanium alloy, TiCode-12 [ASTM grade 12; Ti, 0.3% Mo, 0.8% Ni, 0.2% O_2, impurities of Fe (up to 0.3%), 30 to 50 ppm H, and C], was selected as the prime overpack candidate for further detailed testing. Another titanium alloy, Ti-50A (ASTM grade 2, chemically pure titanium), is considered as a possible backup alloy in applications where temperatures are less than about 100°C (Braithwaite and Molecke, 1980). Both alloys have relatively high costs per unit weight, but Ti-50A is approximately 20% less costly than TiCode-12. The vast majority of our corrosion and metallurgical analyses since 1979 have focused on these two titanium alloys.

Design Alternates. Possible alternative overpack materials to TiCode-12 were also selected, primarily as back-up selections if titanium should prove unacceptable after further detailed testing. These alloys are in the high cost per unit weight category: Inconel 625, Incoloy 825, and Hastelloy C-276; medium cost: ferritic alloys 29-4 (Fe, 29% Cr, 4% Mo) and 29-4-2 (Fe, 29% Cr, 4% Mo, 2% Ni), alloy 6X (50% Fe, 24% Ni, 20% Cr, 6.5% Mo); and, low cost: 2¼ Cr, 1 Mo. This list will be pared to about four alloys after further corrosion screening tests at 150°C in brine A are completed. The second stage of the alternate alloy screening will consist of evaluation of pitting, crevice corrosion, stress corrosion cracking, **and radiation effects and further suitability and modeling analyses, as warranted.**

In support of conceptual waste package designs (for salt repositories), developed by the Westinghouse Advanced Energy Systems Division (1981) for the NWTS program, we included several low-cost cast iron and steel materials in our alternate alloy screening study. Of primary interest are several nodular cast irons and 1018 mild steel. Accelerated corrosion tests are being made in brine A as a function of temperature (50, 70, 90, and 150°C) and time (from 10 d to about 6 months).

General Corrosion

Uniform and Localized Corrosion. The effects of several environmental variables on the corrosion behavior of the titanium alloys and, to a lesser extent, some other alloys of interest have been studied extensively for the past several years. Variables include composition of the corrodent solution (brine A, brine B, seawater, and variations thereof), temperature (25 to 250°C), time (28 d to 6 months), pH (less than 1 to 8), oxygen content (about 30 ppb to 1750 ppm), moisture content (dry, moist, and inundated), and gamma radiation (dose rates of 10^7 and 10^5 rads/h and total doses up to 2×10^{10} rads). Results for dissolved oxygen are listed in Table 5; for effects of temperature, Table 6; and for relative effects of radiation, Table 7.

Most titanium alloy corrosion data were obtained from tests of approximately 30 d and, therefore, represent an average corrosion rate over this period. Recent data on Ti-50A, shown in Fig. 1, indicate that the corrosion rate decreases as a function of time and may, in fact, approach negligible values after about 1 month. The scatter in data is a result of the low corrosion rates and consequent sensitivities to surface preparation. The corrosion rates of TiCode-12 at 250°C in brines and seawater decrease by about 50% over a period of 1 to 6 months. These results suggest that alloy thickness requirements (e.g., for 1000 yr) based on initial corrosion rates may be unduly conservative from the standpoint of uniform corrosion attack.

Both TiCode-12 and Ti-0.2% Pd (an alloy about 40% more expensive than TiCode-12 and, therefore, not as attractive as a candidate) are very corrosion resistant in hot brines and seawater, as is Ti-50A at temperatures of up to about 100°C (Braithwaite and Molecke, 1980). The corrosion resistance is caused by a highly adherent, passivating oxide surface layer. These alloys are more corrosion resistant in oxidizing conditions than in reducing conditions. Generally, conditions that raise the corrosion potential of the metal (e.g., alloying the titanium with nickel) or increase the oxidation potential of the solution so that the alloys stabilize in their passive region are desirable from a corrosion standpoint (Braithwaite, Magnani, and Munford, 1980). Higher temperatures increased the uniform corrosion rate. Figure 2 shows the measured temperature dependence of the corrosion rate of Ti-50A in brine A. These data show

Table 5 Effect of Dissolved Oxygen on the Uniform Corrosion Rate of Titanium Alloys at 250°C*

	Corrosion rate, mm/yr			
	Brine A		Seawater	
Alloy	30 ppb O_2	450 ppm O_2	30 ppb O_2	500 ppm O_2
Ti-50A	0.014	3.2	0.0117	0.0162
TiCode-12	0.0032	0.0018	0.00110	0.00060
Ti-0.2% Pd	0.0024	0.0004	0.00114	0.00062

**Test duration was 30 d.*

Table 6 Effect of Temperature on the Uniform Corrosion Rate of Titanium Alloys in Brine A*

	Corrosion rate, mm/yr		
Alloy	70°C	150°C	250°C
Ti-50A	0.00006	0.0026	0.014
TiCode-12	0.00007	0.0009	0.0032
Ti-0.2% Pd	0.00009	0.00003	0.0024

**Test duration was 30 d; oxygen content, ~30 ppb O_2.*

Table 7 Effect of Gamma Irradiation on Corrosion Rates

Alloy	Dose rate, rads/h	Solution	Test duration, d	Corrosion rate, mm/yr
TiCode-12	10^7	Brine A	49	2×10^{-3}
	10^7	Brine A	87	2×10^{-2}
	10^7	Seawater	49	2×10^{-3}
	10^7	Seawater	87	1×10^{-3}
Stainless steel 304L	10^7	Brine A	79	0.2
	10^5	Brine A	49	9×10^{-4}
	10^5	Seawater	49	7×10^{-4}
1018 Steel	10^7	Brine B	79	1
	10^7	Brine A	79	1
	10^5	Brine A	49	0.1
	10^5	Seawater	49	3×10^{-2}
Inconel 625	10^7	Brine A	87	2×10^{-3}
	10^7	Seawater	87	9×10^{-3}
	10^7	Seawater	49	2×10^{-3}

**Temperature was 90°C.*

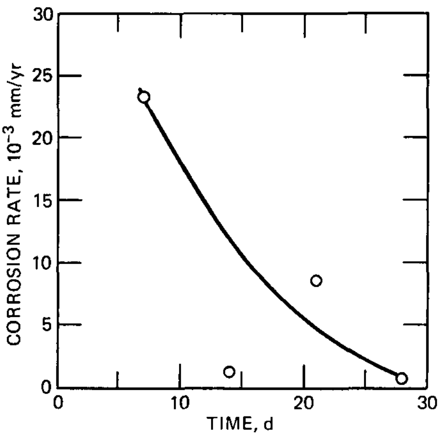

Figure 1 Uniform corrosion of chemically pure titanium vs. time at 250°C in deoxygenated synthetic seawater.

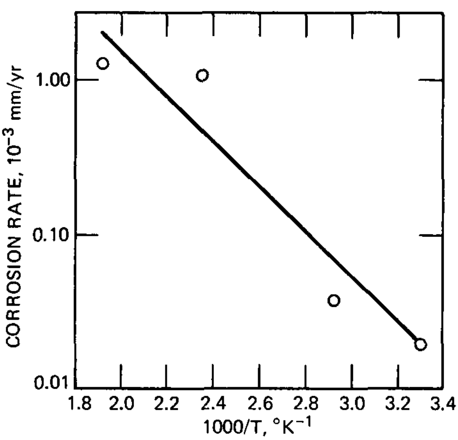

Figure 2 Uniform corrosion of chemically pure titanium, temperature dependence for a 14-d test in deoxygenated synthetic seawater at 250°C.

approximate Arrhenius behavior with an activation energy of 7 kcal/mol.

Localized attack (i.e., pitting) and crevice corrosion were not detected in TiCode-12 or Ti-0.2% Pd, even at temperatures up to 250°C. Crevice corrosion was observed in Ti-50A at 250°C, however; thus it is an unacceptable candidate for high-temperature applications. The probable cause of this crevice corrosion is the development of a low pH (assumed to be ~pH 1) and differential aeration within the crevice as a result of separation of anodic and cathodic reactions. TiCode-12 and Ti-0.2% Pd were developed to avoid crevice corrosion (Braithwaite, Magnani, and Munford, 1980), and they remain stable or passive in the reducing, acid environment of a crevice.

Corrosion Mechanisms. The superior high-temperature corrosion resistance of TiCode-12 to uniform and localized attack in saline solutions is the result of the formation of a highly-adherent, passivating oxide film. We are currently in the process of analyzing this surface layer and evaluating the mechanisms of corrosion protection via several techniques.

1. Electrochemical polarization techniques are being used to define potential regions of active metal dissolution or passivation in various media (particularly strong acids). The use of this technique, as well as other electrochemical methods, will aid in elucidating the corrosion mechanism(s). The initial polarization response of TiCode-12 in brine was described previously (Braithwaite, Magnani, and Munford, 1980). More recent electrochemical results and their implications on mechanisms will be published in the near future (Glass, 1981b).

2. Surface analysis techniques, primarily Auger electron spectroscopy and electron spectroscopy for chemical analysis, will provide elemental and structural information on the oxide films formed in corrosive media.

3. Raman spectroscopy, a similar surface analytical technique, also provides the capacity to study evolving oxide films in an aqueous or gaseous environment.

The role played in corrosion protection by alloy additions to titanium is not well understood. In particular,

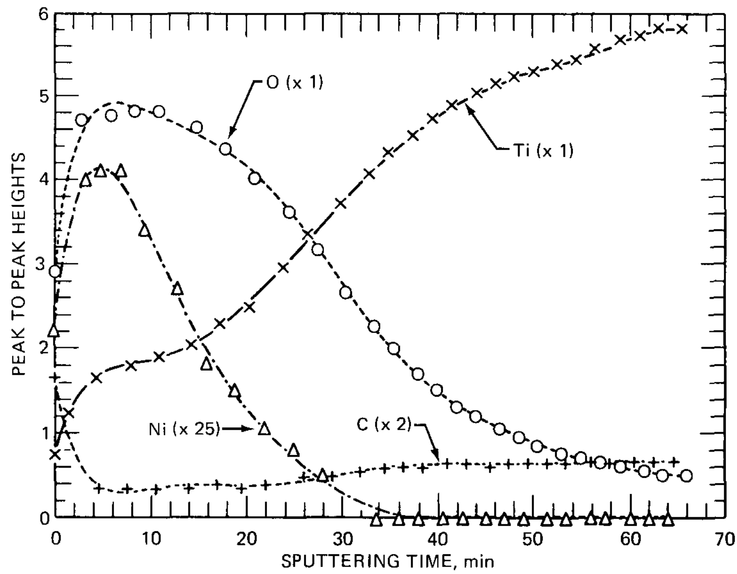

Figure 3 Auger depth profile of TiCode-12 oxidized in brine at 200°C and pH 1.

few analyses have been made of the structure and function of the protective oxide films formed on such alloys under well-defined conditions. A combination of these three techniques is being used to aid in answering such questions.

The influence of nickel and molybdenum alloying additions to titanium in TiCode-12 on the corrosion mechanism(s) is being explored. In particular, the electrochemistry of the binary alloys Ti–Mo and Ti–Ni in corrosive media is being investigated to determine the individual effects of nickel and molybdenum. Possible synergistic effects between nickel and molybdenum may also be important in TiCode-12. Although preliminary, results of such studies indicate that the alloy additions drive the active-passive transition into a more oxidizing region.

An Auger depth profile of TiCode-12 immersed in pH 1, concentrated NaCl brine at 200°C for 3 weeks is shown in Fig. 3. An enhancement of nickel (relative to the bulk metal value) was observed in the oxide film. Molybdenum enhancement or depletion was not detectable because of lack of sensitivity. The observation of nickel enhancement is in agreement with other analyses conducted on other titanium alloys (Glass, 1981b). This enhancement may be responsible, in part, for the corrosion resistance of TiCode-12 by influencing the rate of the cathodic part of the corrosion process and creating a mixed corrosion potential in the passive region or, perhaps, by simply acting as a barrier to anodic titanium dissolution. Such mechanisms remain to be resolved.

Raman spectroscopy of the surface oxide formed on heavily oxidized TiCode-12 specimens (oxidized at 400 to 600°C in air) indicates that the rutile form of TiO_2 had formed. An example spectrum is shown in Fig. 4; the two major peaks correspond to TiO_2 (rutile). Details of the use of this method to examine corrosion products (surface oxides) are given elsewhere (Farrow, Mattern, and Nagelberg, (1979). The technique promises to eluci-

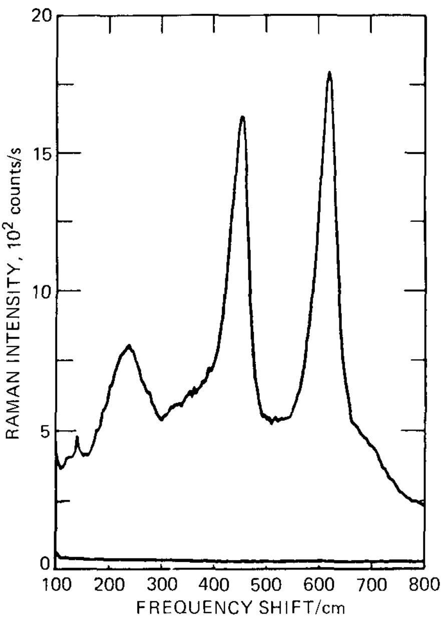

Figure 4 Raman spectra of oxide layer formed on TiCode-12 heavily oxidized in air at 600°C for 72 h.

date corrosion mechanisms. Raman spectroscopy, with the sample immersed in solution is currently being combined with electrochemical polarization to analyze the evolution of oxides formed in different potential regions in a corrosive media. Initial studies of 316L stainless steel indicated the feasibility of such techniques for investigating corrosion mechanisms. Further work in this area is planned. The goal of these studies is to develop a thorough understanding of alloy passivation and a predictive model of the corrosion behavior of TiCode-12 in a salt repository environment. These models will be tested and validated by further laboratory, field, and in situ testing.

Radiation Effects. Effects of high-intensity (10^5 and 10^7 rads/h) gamma irradiation on the uniform corrosion rates of several candidate alloys in seawater are summarized in Table 7. Considerable enhancement of general corrosion was observed under the overtest condition of 10^7 rads/h. In addition, this small amount of data suggests acceleration of the corrosion rates with time, probably because of the buildup of hydrogen in these closed systems. In fact, we have observed hydrogen levels in excess of 10% by volume. At these levels degradation of the protective surface oxides on passive alloys might be expected. A systematic investigation of dose-rate effects and hydrogen embrittlement has been started and is described in the following section.

Glass (1981a) and Dosch (1981) surveyed the literature on the effects of radiation on the chemical environment surrounding waste canisters and the possible effects on the corrosion process. Although the available data are rather limited, results suggest that metals that form adherent, protective oxide films, such as titanium and its alloys, should be least susceptible to the effect on corrosion of radiolysis products. Further data is needed, however, on the diffusion of hydrogen in and through titanium oxide films and its consequent effects on corrosion.

Radiation effects studies now in progress at Sandia National Laboratories, in collaboration with researchers at the University of Minnesota, concentrate on measuring the rate of production of hydrogen from the gamma radiolysis of brines and the resultant effects of this hydrogen on the mechanical properties of titanium alloys. Alterations of the passivating oxide film structure on titanium and titanium-based alloys caused by changes in chemical environment as a result of irradiation in brine are also being investigated.

Environmental Cracking and Embrittlement

An unresolved potential problem in the use of TiCode-12 for overpacks or canisters is that of potential sustained load cracking and stress corrosion cracking and embrittlement caused by a high-temperature brine environment and the presence of hydrogen. Titanium alloys have some hydrogen present in their matrix from manufacturing. They can also absorb hydrogen produced from the radiolysis of brine or groundwater or from possible galvanic reactions with other components of the waste package. The rate of hydrogen absorption is controlled by hydrogen transport through the always present oxide layer.

The susceptibility of TiCode-12 and, to a lesser extent, Ti-50A to stress corrosion cracking (SCC) was assessed by Abrego and Rack (1981) using the slow strain rate technique. In an unnotched material that is susceptible to SCC, a decrease in some measure of ductility (e.g., elongation, ultimate tensile stress, or reduction in area) should be detectable when the material is strained to failure in a hostile environment. Mill-annealed samples were pulled in tension at a constant strain rate (10^{-4} to 10^{-7}/s) until fracture occurred. Environmental test variables included temperature (30 to 250°C), solution (air as reference vs. brine A, brine B, or seawater), pH (1 to 6.5), and dissolved oxygen content (static deoxygenated and flowing oxygen-saturated conditions). Other test variables included prior gamma irradiation in brine, heat treatment, and presence of welds (electron beam and tungsten inert gas welds).

Figure 5 shows a representative sample of some of the results obtained on TiCode-12 in these studies (Abrego and Rack, 1981). In Fig. 5a, the reduction in area (RA), ultimate tensile strength (σ), and elongation (ϵ) in a brine environment are plotted vs. strain rate. In all cases the data are divided by the quantity measured in air. The absence of stress corrosion cracking is suggested by ratios (measurement in brine per measurement in air) near unity. In all cases the material properties appear better in the brine environment than in air, but the differences are within the limits of experimental error.

Even in strongly acidic conditions, TiCode-12 appears resistant to SCC. Data at pH 1 and 3 under oxygenated conditions are shown in Fig. 5b. Ratios of the elongation in seawater to the elongation in air are near unity. In all cases so far, no macroscopic evidence of SCC in TiCode-12 has been observed (Abrego and Rack, 1981).

Alloy composition also plays a major role in determining the degree of SCC susceptibility of predominantly alpha-phase alloys, e.g., TiCode-12 and Ti-50A. Phase structure and phase arrangement tend to be second-order parameters (Blackburn, Feeney, and Beck, 1973). The SCC susceptibility of alpha-phase alloys can also be complicated by the impurity iron content; this iron can stabilize up to about 5% of beta-phase content. On low-temperature aging the beta phase alters to a beta-plus-omega structure, which may lower the threshold stress intensity factor for SCC (K_{ISCC}) for such alloys (Blackburn, Feeney, and Beck, 1973). Presumably TiCode-12 may exhibit similar behavior.

An investigation into the SCC behavior of TiCode-12 under more severe operating conditions (notched, precracked samples) in brine A at 200°C is currently under way. This approach will also be used to establish the influence of alloy chemistry (composition variations) and the processing history of SCC susceptibility of TiCode-12. When a particular chemistry and processing procedure are selected [in a joint program with the Titanium Metals Corporation of America (TIMET)], the SCC behavior as a function of internal hydrogen content will also be evaluated.

Although slow strain rate testing revealed no significant change in

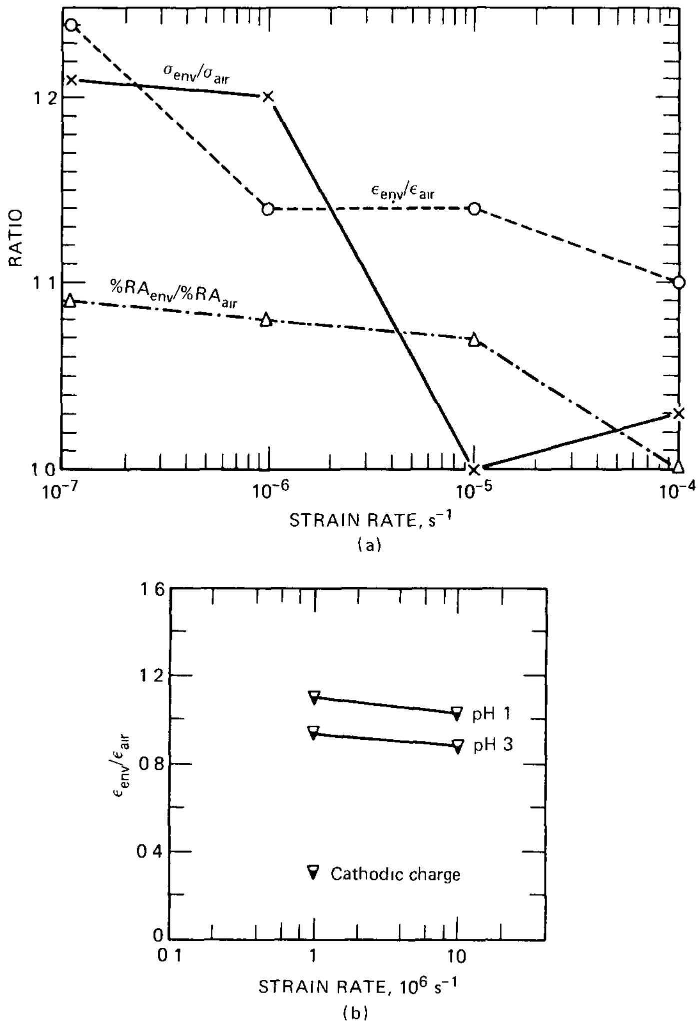

Figure 5 Influence of slow strain rate on stress corrosion susceptibility. (a) Summary of slow strain rate experiments on TiCode-12 in brine A at 200°C. (b) Strain rate dependence of the relative elongation of TiCode-12 in oxygenated seawater at 200°C and pH 1 and 3. The datum of a single cathodically charged sample at room temperature is also shown.

macroscopic tensile properties of TiCode-12, microscopic examination showed that the fracture mode may be sensitive to the test environment (Glass, 1981a). Examination of fracture surfaces (strained at 10^{-7}/s) by scanning electron microscopy yielded some outer-edge quasicleavage fracture regions, as opposed to the ductile-dimple rupture observed in the bulk of the sample. This observation suggests effects caused presumably by hydrogen sorption and embrittlement. A single SCC experiment was performed under strongly cathodic conditions. In this gross overtest case, hydrogen was produced on the base metal surface of the strained sample and severe degradation (stress corrosion cracking) was observed; this is shown in Fig. 5b. The degradation is consistent with the known susceptibility of some titanium alloys to hydrogen embrittlement.

Samples of TiCode-12 strained after gamma irradiation in brine showed no degradation in macroscopic tensile properties. Fractography, however, revealed some evidence of embrittlement in the outer 50 μm of the sample. This observation, plus the results from cathodic charging, makes further study of possible hydrogen embrittlement effects imperative.

TiCode-12 specimens have been charged with hydrogen from a hot gaseous environment and are currently undergoing tensile testing. The effects on mechanical properties of this thermal charging will be compared with those of cathodic charging of hydrogen. The internal hydrogen levels necessary for degradation of mechanical properties and hydride formation will be measured. Other environmental parameters that could significantly affect hydrogen absorption, besides hydrogen input fugacity, include temperature (25 to 250°C), oxide composition and thickness (as a function of pH and exposure time), and the presence of contaminants (e.g., iron) in the oxide.

The measured levels of hydrogen necessary to affect mechanical properties will be compared with the maximum credible levels of atomic and molecular hydrogen produced via brine radiolysis. The microstructural location of the hydrogen (both dissolved and as hydrides) will be identified with tritium tracer and other analytical techniques.

Since the gamma radiolysis of intruding brine is the most significant source of hydrogen in a salt repository, the concentration of atomic and molecular hydrogen produced by radiolysis is being measured in collaboration with researchers at the University of Minnesota. The kinetics of hydrogen species uptake under gamma irradiation, as well as the effect of such irradiation on oxide thickness and composition, will also be measured for several titanium alloys. All these measurements, both with and without irradiation, will assist in our understanding of the mechanisms of potential hydrogen embrittlement and long-term sustained-load crack growth. The effects of hydrogen on titanium alloys need to be evaluated to assess fully their suitability for use as a HLW physical barrier material.

Physical and Mechanical Metallurgy

Laboratory Studies. This segment of the overall canister-overpack program is designed to enable processing of the TiCode-12 or other titanium alloys to provide the best combination of desired properties and to ensure that fabrication methods do not substantially degrade the materials. These studies will include fabrication of large-scale TiCode-12 overpacks for hot-cell and field testing.

A successful overpack material must possess an appropriate combination of mechanical properties, corrosion resistance, and resistance to environmentally assisted cracking (hydrogen embrittlement, sustained-load cracking, and SCC). All these properties of importance are known to be affected by the microstructure and chemistry of the metal. The microstructure may be altered by changes in alloy chemistry, ingot breakdown practice, forming and welding during waste canister-overpack fabrication, final annealing time and temperature, and cooling rate.

Work has been initiated (via a contract with TIMET) to quantify these properties and to optimize chemistry and processing procedures of TiCode-12 in terms of mechanical and corrosion properties. In the first phase of this program, six 70-kg ingots with slightly different compositions were double melted and beta forged into 5-cm-thick slabs. Two final product forms, thin sheet (less than 2 mm thick) and plate (about 6 mm thick), will be investigated. The objective of this work is to be able to write material procurement specifications for TiCode-12.

Previous results (Schutz, 1981) indicated that, when mill-annealed TiCode-12 was reheated and annealed to about 675°C and then corrosion tested in boiling 1 to 2% hydrochloric acid, an increase in corrosion rate occurred. Mill-annealed TiCode-12 consists primarily of equiaxed alpha grains with trace iron in solution and small amounts of beta phase in the matrix and at grain boundaries. The beta phase contains appreciable nickel and iron and a small amount of molybdenum in solution. The presence of Ti_2Ni, converted from the beta phase, has been observed in the aged, sensitized material (Headley, 1981). It is not clear, however, whether this Ti_2Ni is responsible for the increased corrosion rate. This corrosion enhancement is observable only under severe conditions, e.g., in boiling hydrochloric acid solution.

To further study this behavior, Rack et al. (1980) proposed the establishment of time-temperature-transformation diagrams for TiCode-12 and Ti-50A. General corrosion resistance as a function of annealing time, temperature, and resulting microstructure can then be assessed. Currently, all aging experiments have been completed, and corrosion studies in boiling hydrochloric acid have begun. TiCode-12 of various chemistry compositions and binary (Ti-Ni and Ti-Mo) and ternary (Ti-Ni-Mo) alloys are also being investigated to help elucidate the mechanism(s) of enhanced corrosion. This sensitization and enhanced corrosion potential is being

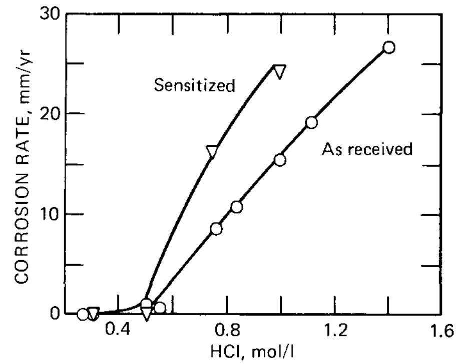

Figure 6 Corrosion of TiCode-12 in strong acid in sensitized (30 min at 680°C) and mill-annealed conditions.

investigated in detail because of the impact it may have on weld-affected zones of an overpack.

Figure 6 compares the pH dependence of the corrosion rate for sensitized vs. unsensitized (mill-annealed) TiCode-12. The data indicate increased susceptibility to general corrosion in strong acids (pH values less than 1) for sensitized specimens but show no significant enhancement at credible pH values. Current effort is directed at the "knee" of the curve since small shifts in the threshold for enhanced corrosion may lead to crevice attack.

Overpack Thickness and Installation. The thickness of TiCode-12 required for fabricating an overpack is based on the requirements of corrosion resistance and mechanical considerations. On the basis of the data presented so far, a thickness of 1 to 2 mm appears to be quite adequate to provide corrosion-resistant barrier integrity for 1000 yr or more in the expected environment of a salt waste repository. The overpack must also be able to survive the mechanical rigors of handling and emplacement operations in the repository, however. It must be capable of adequately withstanding the crushing and deformation stresses exerted on it by the lithostatic pressure in a salt repository without suffering damage to welds or seams by nonuniform compression, wrinkling, etc. For these reasons an overpack thickness of up to 6 mm, assumed conservative, has been proposed (Westinghouse Advanced Energy Systems Division, 1981). The actual thickness of the overpack must reasonably balance the required mechanical properties vs. costs so that it will not unnecessarily drive up the total cost of this barrier.

The mechanical vs. materials (cost) requirement can be addressed in several ways:

1. The overpack can be applied to the waste canister before filling (e.g., with molten glass); thus there is no gap between these two physical barriers. Lithostatic pressures applied to the outside of the waste package will be transmitted to the solid, self-supporting wasteform. This concept requires that any void volume within the canister (e.g., the empty top 10 to 15%) be structurally reinforced internally (before filling) or remotely post-filled with a self-supporting material (metal beads, sand, glass frit, marbles, etc.).

2. The canister can be fabricated of TiCode-12 or other corrosion-resistant alloy (e.g., Inconel 625, Incoloy 825, or Hastelloy C-276), thus eliminating both the stainless steel and the separate overpack. Any non-glass-filled void within the canister would have to be structurally reinforced internally or postfilled to support lithostatic pressures.

These two options for defense HLW packages are currently being evaluated at Sandia National Laboratories in conjunction with personnel from Savannah River Laboratory.

Because of fabrication tolerances (out of roundness) of a stainless steel 304L waste canister and canister deformations (localized bulging or nonlinearity over the length of the canister) caused by nonuniform heating during waste filling, a gap of about 12 to 25 mm may be necessary between the canister and the overpack. This would allow the canister to be inserted into the overpack without significant risk of binding. For heat transfer and structural support reasons, it is desirable to minimize or eliminate this gap. This leads to other canister-overpack options.

3. A thin (2- to 3-mm-thick) TiCode-12 overpack could be fabricated around a thick (about 10-cm) sleeve or structural "overpack support." The glass-filled waste canister (with no internal structural reinforcement) could then be simply inserted into the TiCode-12-clad overpack support and sealed. The purpose of the thick structural overpack support is to prevent crushing by lithostatic pressure) of the top 10 to 15% void volume in the canister. This is the present conceptual design reference option for the borehole concept waste package (Westinghouse Advanced Energy Systems Division, 1981).

4. An overpack-to-canister gap could be filled with a castable solid material, e.g., a lead or bismuth alloy filler, so that lithostatic pressures could be transmitted to the solid, self-supporting wasteform.

All these options are being evaluated to resolve the uncertainties associated with them. Issues to be resolved include cost of materials (and minimization thereof), materials behavior, techniques and ease of fabrication, overall size of waste canisters and overpacks, and impacts on waste processing and filling facilities. Studies at Sandia focus on the first two of these issues.

Advanced Hot-Cell and Field Testing

Several advanced large-scale tests are in progress now or are in the planning stage: the Battelle Northwest Laboratory (PNL)-Sandia National Laboratories HLW package interactions test, the Institut fur Tieflagerung/Federal Republic of Germany (IFT/FRG)-Sandia cooperative test in the Asse salt mine facility, and the waste package materials field test being conducted in a halite region of a potash mine in southeastern New Mexico. The purpose of these various tests is to evaluate the performance of a TiCode-12 overpack under salt mine conditions in conjunction or interaction with other waste package barrier components.

PNL–Sandia HLW Package Interactions Test. The first stage of the HLW package interactions test has been completed (Molecke, Bradley, and Shade, 1981). This test involved all components of a conceptual waste package to be used in a salt repository; they included reprocessed HLW glass (PNL 76-68, 7.6 cm in diameter by 38 cm long, loaded with inactive fission products and ^{238}U), a stainless steel 304L canister, a TiCode-12 overpack, a bentonite-sand backfill, and a machined, bedded rock salt container—all within a 19-l autoclave. To force wasteform-barrier-salt interactions to occur during this first test (95 d at 250°C), we compromised all barriers physically, as shown in Fig. 7. Excess concentrated brine was intentionally added to the total system (with the backfill mixture, as a slurry) to accelerate waste form leaching or alteration and allow analysis of their extent, and to measure radionuclide release and retention on other barrier components.

This experiment, conducted jointly by Battelle Pacific Northwest Laboratory and Sandia National Laboratories (Bradley, Shade, and Molecke, 1980), is a part of the NWTS HLW experimental program. This test is the first moderate-scale experiment involving all suggested waste package components, although in a compromised overtest manner. Its purpose was to gather data on how the various barriers act in concert to retard radionuclide release from the total package. Data obtained from this complex, multi-component test are presented in detail elsewhere (Bradley, Shade, and Molecke, 1980; Molecke, Bradley, and Shade, 1981) and are compared with accelerated test data from previous single- and double-component tests (Molecke, Bradley, and Shade, 1981) conducted at either PNL or Sandia. The data obtained will be used to help predict interactions and performance of the various waste package barriers under realistic salt repository environmental conditions.

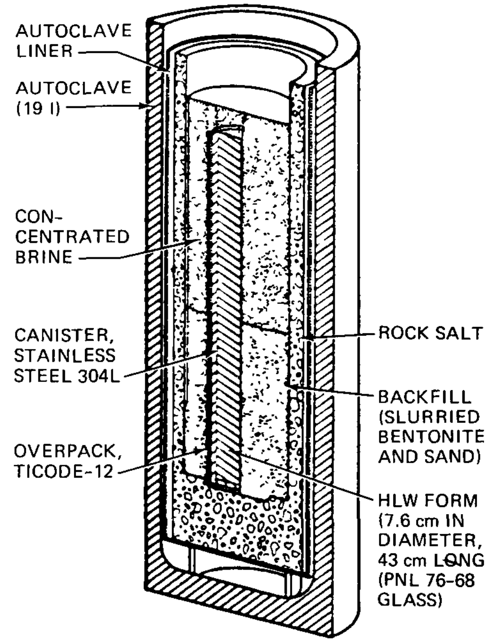

Figure 7 Joint PNL–Sandia HLW package interactions test.

The next phase of the PNL-Sandia HLW package interactions test will involve the use of a high-intensity gamma irradiation source and credible salt-repository environments (a maximum temperature of 150°C). The

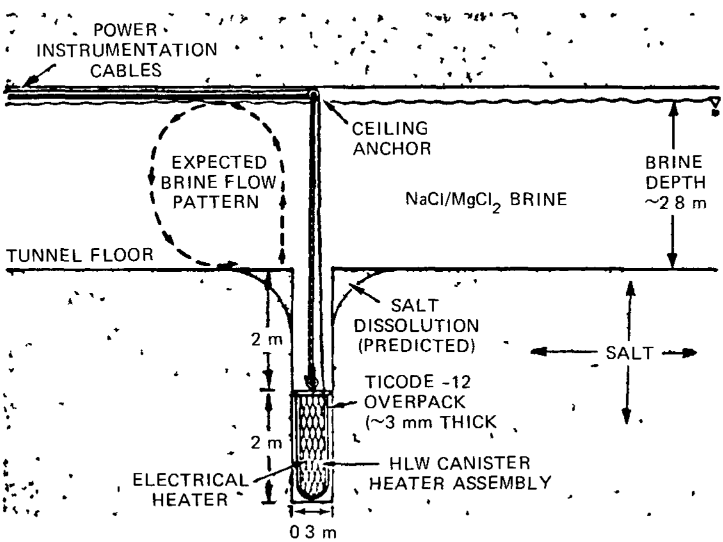

Figure 8 IFT/FRG–Sandia cooperative test in Asse.

final phase will incorporate radioactive, fully loaded PNL 76-68 glass. Equipment is being prepared and experimental techniques are being developed for these tests (Molecke, Bradley, and Shade, 1981).

IFT/FRG–Sandia Cooperative Test in Asse. Preliminary planning of a joint IFT-Sandia in-mine corrosion test of a TiCode-12 overpack in the Asse salt mine facility in the FRG has been completed This test involves placing a 2-m-long by about 0.3-m-diameter TiCode-12 overpack, surrounding a steel heater (HLW canister simulant), into a borehole in the floor of a salt drift (Fig. 8); the mine room will be flooded with saturated brine during the test. This test is scheduled to be emplaced in 1982. Sandia will design and fabricate the overpack, using 3 2-mm TiCode-12 sheet. The initial test design is a joint IFT-Sandia effort, as will be the post-test analyses of the overpack.

Waste Package Materials Testing in Southeastern New Mexico Salt. This series of in-mine tests consists primarily of chemical and physical tests on emplaced HLW backfill barrier materials and, to a lesser extent, corrosion tests of candidate HLW canister-overpack materials. An objective of these tests is to compare in-mine field data with previously obtained laboratory results, particularly on overpack-backfill interactions

Samples of selected candidate alloys are attached to the electrical heater (HLW canister simulant). The heater jacket is constructed of 10-cm-diameter Inconel 600 pipe about 1 m long. The alloy coupons will be galvanically isolated from each other and in direct contact with dry or moist backfill material (bentonite clay or a bentonite-charcoal-sand mixture) at either 150 or 250°C. Post-test analyses will be conducted to measure general corrosion rates and to look for indications of localized attack (pitting and crevice corrosion). The first of these tests started in December 1981. Six tests are planned, each lasting 2 to 3 months

Summary

Approximately 20 candidate alloys have been examined for corrosion resistance under environmental conditions representing a salt waste repository and also under the overtest conditions. On the basis of a series of corrosion, metallurgical, and economic analyses, the titanium alloy TiCode-12 was chosen as the prime candidate for an overpack barrier that may survive intact for 1000 yr or more. TiCode-12 has not been disqualified by any of the detailed analyses and overtests conducted as part of our program. These analyses include evaluation for uniform corrosion, localized attack, crevice corrosion, environmental cracking, and embrittlement; response to gamma irradiation in brine; and sensitization. The current status of these studies is described in this paper. Further analyses of the applicable corrosion mechanisms and degradation modes, e.g., sustained load cracking as a result of hydrogen embrittlement, are still in progress.

Other materials are also currently being evaluated for uniform and localized corrosion for possible use as backup or design alternate overpacks under representative salt repository conditions. These alloys include Inconel 625, Incoloy 825, Hastelloy C-276, ferritic alloys 29-4 and 29-4-2, Alloy 6X, 2¼ Cr–1 Mo steel, 1018 mild steel, and several cast irons.

The overall Sandia HLW canister-overpack program is moving away from predominantly laboratory testing and more toward concerns of overpack metallurgical and structural response under repository conditions, concerns of waste package design and fabrication, and advanced hot-cell, field, and, ultimately, in situ testing and demonstrations.

Acknowledgments

Appreciation is expressed to former co-workers J. W. Braithwaite, L. Abrego, and H. J. Rack for their many past contributions. Some of their data and analyses, all in support of the Sandia HLW canister–overpack program, have been incorporated into this paper.

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Waste Package Performance Analysis

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A performance assessment model for multiple barrier packages containing unreprocessed spent fuel has been applied to several package designs. The resulting preliminary assessments were intended for use in making decisions about package development programs. A computer model called BARIER estimates the package life and subsequent rate of release of selected nuclides. The model accounts for temperature, pressure (and resulting stresses), bulk and localized corrosion, and nuclide retardation by the backfill after water intrusion into the waste form. The assessment model assumes a post-closure, flooded, geologic repository. Calculations indicated that, within the bounds of model assumptions, packages could last for several hundred years. Intact backfills of appropriate design may be capable of nuclide release delay times on the order of 10^7 yr for uranium, plutonium, and americium.

Introduction

A waste package performance assessment model has been applied to typical multiple-barrier package designs for spent fuel intended for geologic disposal. A computer model called BARIER was developed to evaluate the performance of proposed design concepts, assess the sensitivity of package performance to specific parameters, and support the evaluation of incentives for use of various designs.

The model was developed for specific circumstances and was, therefore, simplified for limited application, but it is structured in subprograms to allow expansion and improvement for future applications. A specific list of materials was considered (others can be added) as applied to a specific design based on previous conceptual studies (Westerman et al., 1979). The assessment model assumes that the repository is flooded (post-recharge).

Model Description

The performance assessment model is a combination of finite difference and analytical solutions. The package failure model, which calculates time to water intrusion on the waste form, is a time-stepped marching solution. The nuclide release model incorporates analytical solutions to nuclide transport equations to estimate nuclide release rate vs. time after the package begins leaking. Detailed information on the model is discussed in Stula et al. (1980a, 1980b).

The package is viewed as a multilayered (multiple barrier) assembly, which undergoes a failure process starting with the outermost barrier and proceeding inward. Each barrier element is envisaged as shown in Fig. 1, and together the elements form a package, as shown in Fig. 2. (Note that the number of barriers may be less than or more than that

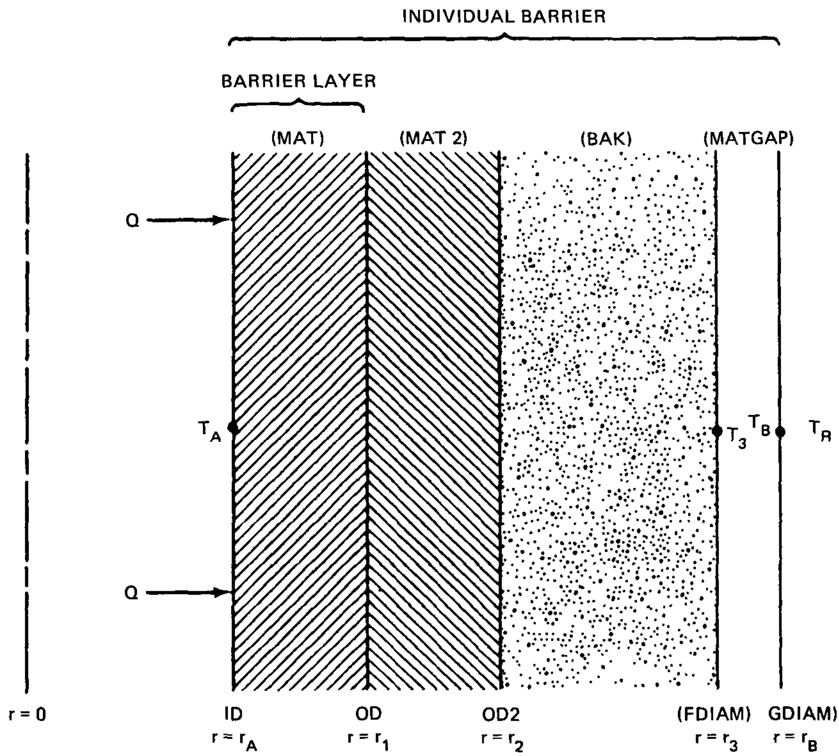

Figure 1 PKTEMP barrier model; where Q is the radial waste heat generation (watts); T_A is the temperature at the inside of the leftmost solid wall, or waste (°K); T_B is the temperature at the outside of the rightmost solid wall, or repository (°K); T_3 is the temperature at the outside of the backfill, or filler (°K); r is the radius relative to waste centerline, r = 0 (in.); (MAT), (MAT 2), (BAK), and (MATGAP) are dimensional or material variable names used in the code; and T_R is the temperature of the repository (°K).

shown in Fig. 2.) The outer material (MAT 2) of a barrier is assumed to possess no structural strength and to act only as a corrosion protector or radiation shield. The existence of solid walls, fillers, or gaps in a particular design is conveyed to the model by setting the diameters of each barrier layer to the appropriate value. If a particular barrier layer does not exist, then the inside diameter of that layer is set equal to the outside diameter. The inner barriers are protected from corrosive attack and from external forces by the outer barriers. As each barrier fails, the next inner barrier is subjected to the water environment and the pressure and temperature conditions of the repository.

Figure 3 shows how the model assesses the successive failure and attack of the barriers, which lead to leaching and radionuclide release after failure of the last barrier. Initially, a heat transfer model is used to determine the maximum steady-state temperature that the waste would attain if the package remained intact in a repository at its maximum temperature. If a temperature of 653°K (380°C) is attained in the fuel bundle, the package is rejected, and no further

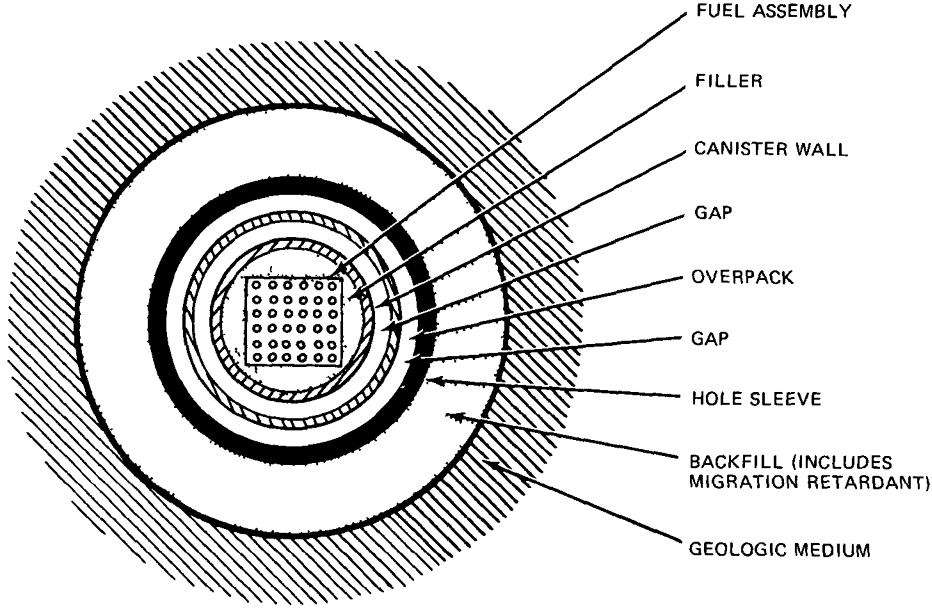

Figure 2 Stylized waste package configuration for fuel assembly waste form.

calculations are made. If the temperature is within limits, the package is taken through time increments, as shown in Fig. 3. The temperature of the outer barrier is assessed in the heat transfer model, and the nuclear radiation field is evaluated. Then the corrosion model determines the decrease in barrier wall thickness for that time increment based on the water chemistry, type of material, and temperature range. The revised wall thickness is checked in a mechanical stress model that calculates displacement and stresses and checks the results against failure criteria. If the element does not fail, the time is incremented and the process is repeated. When the barrier fails, the next innermost barrier is taken through the process until the last barrier fails. This final failure passes control to the waste package release model, which includes leaching and transport calculations for specific radionuclides.

Temperature Calculations. Temperatures are calculated using areal heat loadings assumed for the reference waste repository as described in the generic environmental impact statement (GEIS) (Department of Energy, 1979). It is conservatively assumed that the bulk temperatures are unchanged by the presence of water from the flooding scenario. An approximate fit to the time-temperature curves in the GEIS is made for each of the four geologic media considered.

A concentric cylinder model is used which accounts for heat transfer by conduction and radiation. Exploratory calculations revealed that free convection effects are small and that coefficients tend to approach pure conduction. When the waste package has gaps between barrier elements, heat radiation effects are included in the model.

Corrosion. The corrosion subroutine calculates the thicknesses of the

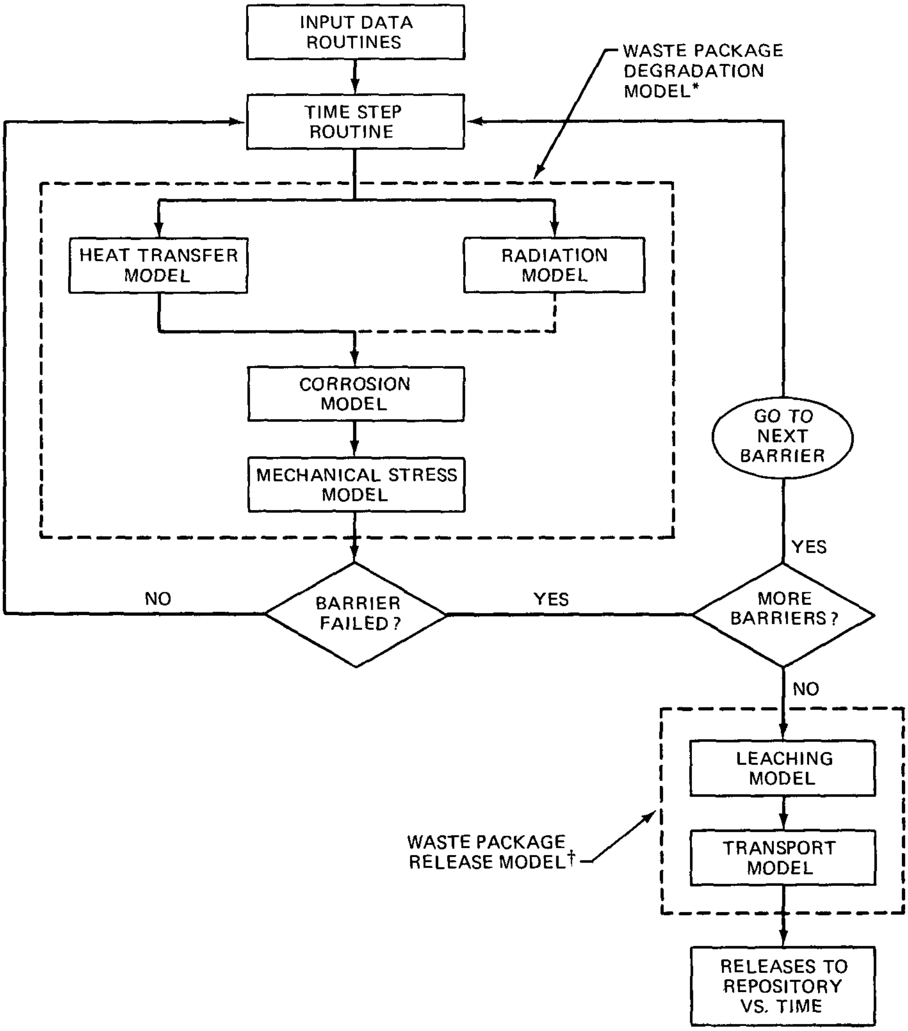

Figure 3 Preliminary waste package degradation and release model.

****The waste package degradation model describes the processes that lead to the contact of the waste by transport agents.***

†The waste package release model describes the processes of radionuclide releases from the waste forms and subsequent transport through the migration retardant to the repository.

two inner layers of each barrier as a function of time. In each case a corrosion rate, chosen on the basis of the temperature and type of repository water, is used to calculate the decreasing thickness of a solid barrier wall. The model assumes that the corrosion rate is characteristic of full immersion conditions.

The corrosion calculations are performed according to the logic shown in the flow chart in Fig. 4. The program first tests for the existence of a corrosion-resistant coating on the out-

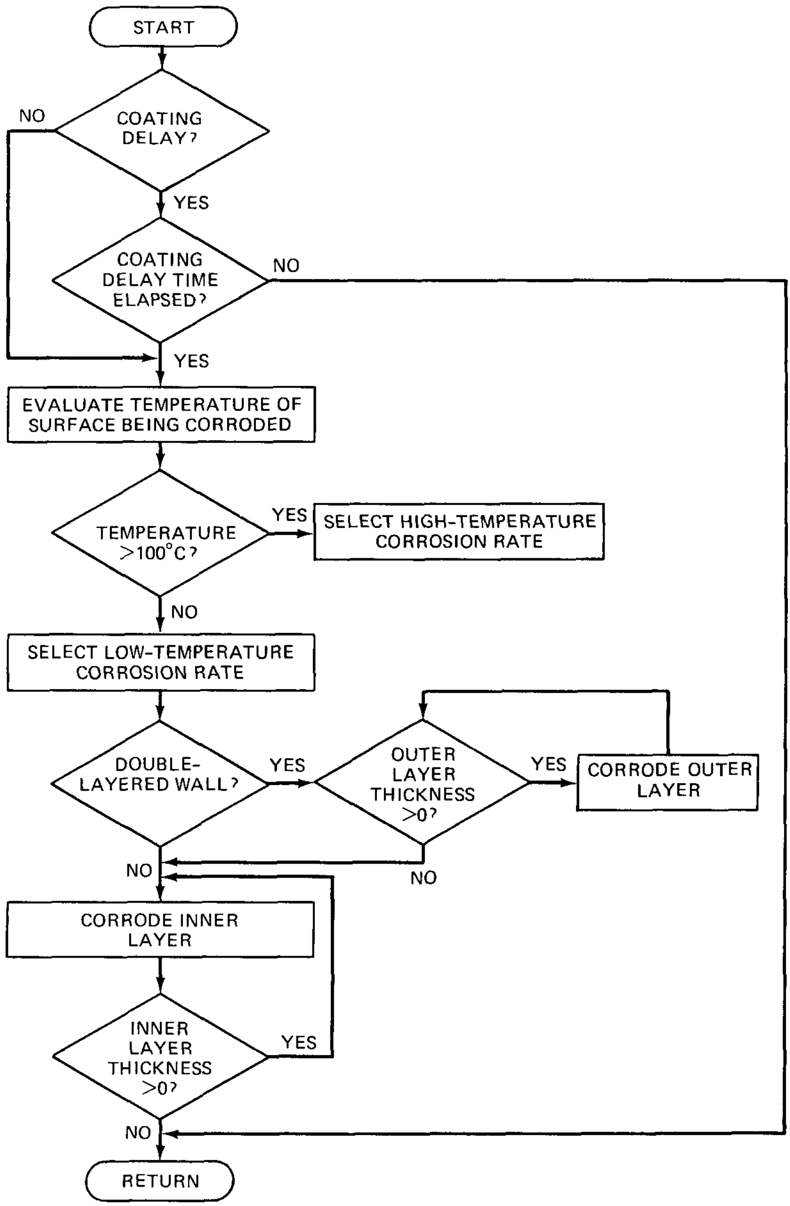

Figure 4 CORODE flow chart.

side of the outermost of the two barrier layers in question. If present, the coating is defined in terms of time of protection afforded to the surface to be corroded. This length of time is specified in the input data files. Calculations do not begin until the specified time period has elapsed. Corrosion rates are chosen on the basis of the temperature calculated and the existing repository water type. For each pass through the subroutine, the outer of the two layers in question is decreased in thickness by an amount equal to the corrosion rate times a time increment (specified in input to the main BARIER program) until cladding layer thickness is zero. After failure of the outer layer, the inner layer is corroded at the appropriate rate until it fails because of either zero thickness or excessive stress. When a complete barrier fails, the next innermost barrier is considered to be uniformly flooded, and the entire process is repeated.

The corrosion mechanisms considered include uniform corrosion, stress corrosion, pitting, and graphitization. Package materials for which corrosion data are obtained include mild steel, Zircaloy-2, Iconel 600, stainless steel 304, copper lead, and cast iron.

The corrosion rate data are somewhat uncertain because of the numerous effects of environmental parameters on package corrosion. Environmental parameters acting on waste packages vary with the geology of the repository and can have a major impact on the resulting corrosion rates. For example, increases in temperature generally increase the corrosion rates of metals (Braithwaite and Molecke, 1979). Also, increases in temperature in an open system cause a depletion of dissolved oxygen in aqueous solutions. This decreases the corrosion rate of metals whose rate is controlled by diffusion of oxygen.

The restraining pressure to which a waste package is subjected in a repository affects the corrosion rate primarily in that it influences the physical state of intruding water and the concentration of dissolved gaseous species. Waste packages will be exposed to any thermal decomposition products of the geologic isolation formation and any dissolved and gaseous species present. In general, species in solution that increase the oxidizing power of the solution increase the corrosion rate.

The tensile stress present in the barrier wall is an essential factor in stress corrosion cracking. Not all materials are susceptible to stress corrosion cracking in geologic isolation conditions. For those which are, the threshold tensile stress depends strongly on temperature, solution composition, and the presence of an aqueous phase. Alloys containing carbon and chromium can be susceptible to sensitization. For example, sensitization in stainless steels refers to the thermally induced formation of chromium carbide at or near grain boundaries (Braithwaite and Molecke, 1979). This increases the susceptibility of the alloy to intergranular attack and intergranular stress corrosion cracking. Welding, because of the high temperatures involved, often leads to sensitization and tensile stress in welded regions. The corrosion rate data base is generally considered to be conservative in view of the uncertainty about the physical conditions expected in a repository.

Failure Criteria. The barrier is considered a bimetallic wall adjacent to a porous filler (or backfill as shown in Fig. 5). The model assumes a structural wall, a cladding with no strength attributes, and a structural backfill. Stress-strain properties of the backfill and the structural wall determine pressure profiles between R_0 and R_1 and R_2 and R_3. The pressure is assumed uniform between R_1 and R_2.

A stress equilibrium calculation is used to determine the response of the cylindrical composite at a stressed state in comparison with pressures and dimensions in an unstressed state. Because changes occur very slowly, it is reasonable to assume equilibrium.

Metal barrier walls are considered to fail as a result of external pressure when the wall is in plastic strain and there is a uniform pressure across it ("hydrostatic"). Failure caused by internal pressure occurs when the

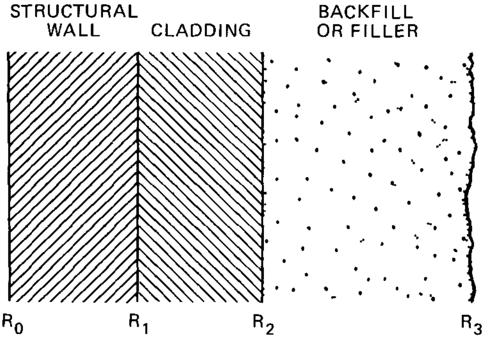

Figure 5 Composite barrier used in stress calculations.

wall thickness no longer meets the requirements for hoop stress in the American Society of Mechanical Engineers code (Section VIII, Division 1). Wall thickness is the portion of the original wall that is not affected by corrosion (including bulk corrosion, pitting, or crack propagation) as determined in the corrosion subroutine. The subroutine updates two binary flags, BFAIL and WFAIL. If BFAIL is 0, then the backfill has "failed"; this means there is no longer a pressure gradient across the backfill. If WFAIL is 0, the solid wall has failed. If WFAIL or BFAIL is 1, the wall and backfill are intact, sustaining a pressure gradient.

In each time increment the wall thickness and temperature of a barrier are revised. Then the failure subroutine recalculates the new stress distribution, and the main program determines when the defense shifts to the next inner package barrier.

Radionuclide Release Rates. The radionuclide release model calculates the transport rate of specific radionuclides through failed engineered barriers and backfill. The specific rate of interest is the release rate to the geology. The model is based on slab geometry, a conservative assumption relative to a cylindrical geometry. The engineered barrier package can consist of many layers of different materials. At some time after emplacement in the repository the barriers fail, either by crushing from the lithostatic pressure in the repository or by corrosion-caused leaks. In either case, when the barriers fail, it is assumed that water is available throughout the fuel bundle, barriers, and backfill and that mass transport by diffusion begins.

The radionuclide release model calculates the release rate based on Fick's second law of diffusion, i.e., no countercurrent diffusion and no convection of water. The backfill is assumed to have capacitance in excess of that of a solution. The capacitance is caused by sorption of the species of interest. Resistance to mass transfer is also assumed to exist because of the remains of failed barriers. This assumption is reasonable because there is a finite distance from the waste to the backfill face, and the failed barriers represent a physical resistance through a void fraction available for transport, i.e., a porous barrier. The failed engineered barrier is assumed to have no capacitance since the capacitance of the backfill is much larger. At the beginning of release, the concentration of nuclide in the canister water is assumed to be the equilibrium value. This is valid because diffusion out of the package is expected to occur at rates many times slower than dissolution rates. An analytical solution to the diffusion equation is used to calculate the nuclide concentration profiles through the package and the resulting release rates to the repository as a function of time.

Nuclear Radiation Fields. The radiation field subroutine calculates the radiation exposure from gamma rays at the outer surface of each package barrier as a function of time. The radiation source in the model is assumed to be 3.3% enriched spent

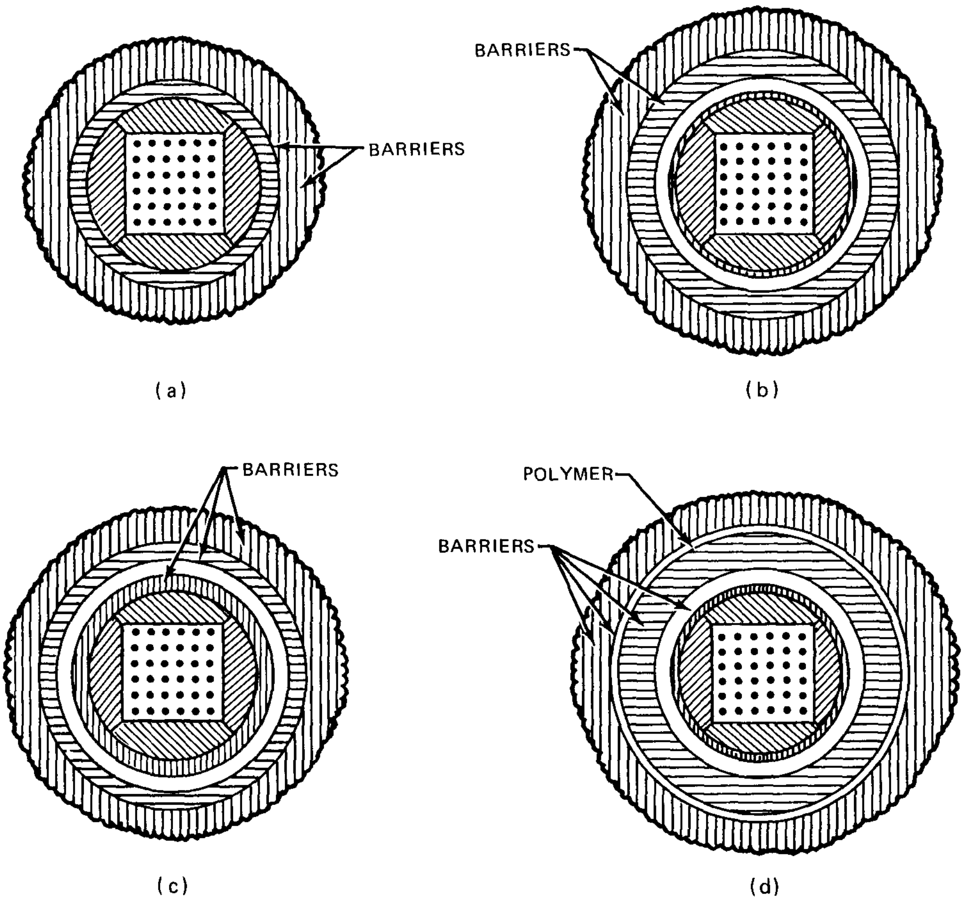

Figure 6 Conceptual designs of barrier system from SURF/SFHP program. (a) Corrosion-resistant metal canister in specially tailored backfill; the maximum number of barriers is 2. (b) Mild steel canister with corrosion-resistant overpack or hole sleeve in specially tailored backfill; the maximum number of barriers is 2. (c) Corrosion-resistant canister with corrosion-resistant overpack or hole sleeve in specially tailored backfill; the maximum number of barriers is 3. (d) Mild steel canister in thick corrosion-resistant bore sleeve with surrounding polymer layer in specially tailored backfill; the maximum number of barriers is 3, or with corrosion-resistant canister, 4.

fuel from a pressurized water reactor, with a burnup of 33,000 MWd/tonne. Emplacement time is assumed to be 6.5 yr after discharge, and the burnup is assumed to be constant over 1100 d. Various materials are chosen for use in the engineered barriers of a package design. Compositions and densities of some of these materials are obtained to estimate the radiation attenuation characteristics. The data for the other materials are estimated or assumed. Data for a shielded cylindrical source are calculated with a buildup factor approach (Rockwell, 1956).

Applications

The performance evaluation model has been applied to a large variety of package design variations (several hundred). Most of the package designs that have been evaluated were based on the concepts described in the spent unreprocessed fuel (SURF) program (Westerman et al., 1979).

Variations on four basic SURF program concepts have been considered (Fig. 6) and are designated A, B, C, and D. An additional concept (E), in which the stabilizer was a cast-in-place solid rather than segmented blocks, was studied. Details of all designs studied are found in project reports (Stula et al., 1980b; Lester et al., 1979).

Table 1 Ranges of Package Performance for Typical Design

Rock and package types	Time to waste–water contact, yr
Salt	
A	5
B	1,000
C	20
D	100
E	2,500
Shale	
A	820
B	1,000
C	100
D	200
E	13,000
Basalt	
A	10,000
B	10,000
C	20,000
D	13,000

Results

Package Performance Evaluation. Detailed results for hundreds of package design variations have been generated and reported (Stula et al., 1980b; Lester et al., 1979, 1980). Table 1 summarizes results obtained for the best examples of each package type described in the previous section.

The best A packages are generally constructed of stainless steel 304 or Zircaloy, with walls 0.64 cm (0.24 in.) thick. Concept A packages do not perform well in media with high creep stress since there is no heavy-walled sleeve or rigid waste form. This package does well in hard rocks such as basalt.

The best B packages are those with a Zircaloy cladding on a heavy iron (32-cm, 13-in.) sleeve for use in high creep media. In hard rock a Zircaloy overpack of 4.4 cm (1.75 in.) works as well as the sleeve. In hard rock a heavy lead (sacrificial) sleeve cladding also extends package life.

Concept C packages use a corrosion-resistant canister with a sleeve or overpack and generally combine the features discussed under B packages. Except for the use of pure Zircaloy in hard rock, the C packages do not offer any special advantage.

Concept D packages perform well in hard rock because high corrosion-resistant materials like Zircaloy or stainless steel 304 are used.

Concept E packages display superior performance in the high creep media since the solid waste form helps the entire package to resist failure by mechanical crushing. The E packages evaluated incorporated a cast lead stabilizer and a Zircaloy or Zircaloy-clad stainless steel can 7.6 to 25.4 cm (3 to 10 in.) thick.

In all designs, sand–bentonite backfill mixtures were found to delay plutonium release by up to 10^4 to 10^5 yr and to spread the release over a period of 10^5 to 10^7 yr. Results of a similar order of magnitude were obtained for uranium. Generally, americium is attenuated large enough that no release can be calculated. Results for americium are difficult to obtain because the inventory is too small.

Sensitivity Studies. Sensitivity analyses have been performed to determine the effects of certain physical characteristics and geologic conditions on package performance. The effects of variations in repository temperature and pressure, waste heat generation rate, gap thickness between package barriers, backfill thickness and compaction coefficients, and radionuclide solubility were evaluated.

Repository pressure was found to have no effect on designs using a cast stabilizer. However, in designs with non-cast stabilizers, canister thickness at failure and hence leach begin time are affected significantly. As repository pressure increases, canister thickness required to withstand media creep forces increases and leach begin time (or time of canister failure) decreases.

In all cases, peak waste temperature is affected only to the extent that repository temperature varies. That is, the temperature gradient between repository and waste is constant and depends on waste heat generation rate. Repository temperature was found to have a small but significant effect on canister thickness at failure for the non-cast stabilizer designs. The criteria used to determine minimum canister thickness required to withstand geologic creep forces are temperature dependent. Thus leach begin time is inversely related to canister thickness at failure. For cast stabilizer designs, no effects on canister thickness at failure or leach begin time are evident. According to the BARIER corrosion model, temperature would affect corrosion rate to the extent that one of two corrosion rates corresponding to two temperature ranges would be used in any particular corrosion calculation.

The maximum waste temperature increases linearly with increasing waste heat generation rate. This is to be expected from the nature of the equation for heat transfer by conduction only. Conduction and radiation equations give a linear dependence of maximum waste temperature on waste heat generation rate. This indicates that for relatively small air gap thicknesses within a package, the radiation component of heat transfer is of minor importance in comparison to the conduction component.

Varying backfill compaction coefficients was found to have no effect on package life or any other performance characteristic with the exception of net pressure on a barrier at failure. However, this effect is relatively minor over the range of compaction coefficients considered. For cast stabilizer designs, the net pressure of a barrier at failure is independent of backfill compaction coefficients.

Most of the radionuclide transport resistance, as calculated by the release subroutine, is attributed to the backfill thickness, except when the backfill thickness is extremely small [less than 2.5 cm (1 in.)]. Radionuclide release rates reach steady state more quickly as the backfill thickness is decreased.

The effects of solubility of ^{238}U on radionuclide release rate are evident in the results of each package design evaluated. For the high solubility case, the release rate reaches steady state more quickly and is significantly higher than in the low solubility case.

Conclusions

Even though the results obtained are based on highly conservative assumptions, some possible synergistic effects (e.g., between stress and corrosion) are not accounted for. Also some effects, such as nonuniform stress (which could shear the package in two), could not be modeled with data currently available. Thus caution must be observed in attempting to apply the performance evaluation model or any of the reported results. All the details contained in project reports referenced in this paper should be carefully studied and considered in any further applications.

The predicted performance of the packages studied thus far indicates that multiple engineered barrier packages can be designed to last several hundred years beyond closure of the repository. If the repository remains dry, much longer package life can be expected. Subsequent nuclide release could be distributed over a time period of the order of 10^7 yr assuming that the backfill remains intact and exhibits reasonably good sorption properties over such a time period. Some question of backfill lifetime still remains, especially in relation to permeability and sorption properties.

The BARIER code provides preliminary performance estimates for use in making design decisions during the early stages of package and repository system design. The model is currently being further developed and improved. An improved model will be available at a later date. Further use of the model to evaluate later designs is expected.

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